NPP LOVIISA-3

Expert Statement to the EIA Report

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ÖSTERREICHISCHES ÖKOLOGIE INSTITUT

umweltbundesamt

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CONTENT

1 INTRODUCTION

The company Fortum Power and Heat Oy (Fortum) plans to construct a new nuclear power plant (NPP) at the island of Hästholmen in Loviisa. Loviisa is the location of two operating NPP units. Electric capacity of the third NPP Unit shall be 1,000 to 1,800 MWe.

According to the Finnish law the construction of a new nuclear power plant is subject to a decision-in-principle[1] issued by the Government and ratified by the Parliament.

With reference to the ESPOO Convention the Austrian Federal Ministry of Agriculture and Forestry, Environment and Water Management, has expressed its interest to take part in the transboundary EIA. The Austrian Institute of Ecology was assigned by the Austrian Ministry of Agriculture and Forestry, Environment and Water Management to elaborate an Expert Statement on the EIA Program for Loviisa-3 (LO 3) NPP. In the second stage of the EIA process the Austrian Institute of Ecology in cooperation with Dr. Helmut Hirsch was engaged by the Austrian Federal Environmental Agency to assess the Environmental Impact Assessment Report of Fortum.

The findings of this evaluation are presented in this Expert Statement, which is structured as follows:

Chapter 2 presents the summary of the Expert Statement; Chapter 3 discusses the EIA procedure. Chapter 4 deals with the reactors considered for LO 3. Chapter 5 concerns safety and accident analysis. Chapter 6 and 7 concern the environmental impact of nuclear fuel production and the repository of irradiated fuel. Chapter 8 contains a short evaluation of the information given by the EIA Report on alternative options. Questions are summarized in chapter 9.

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[1] now: favourable resolution

2 SUMMARY

The project under discussion seems not to be elaborated well enough to fulfil the requirements of the EC EIA Directive (EC 97/11) and the ESPOO Convention (ESPOO Convention 1997), because the EIA Report does not present a certain project and alternatives. In the EIA Report, Fortum examines the construction of a nuclear power plant unit with an approximate net electrical output of 1,000 to 1,800 MW and thermal power of 2,800 to 4,600 MW at Loviisa. The reactor type could be a BWR or a PWR. A list of ten reactor types is presented, but Fortum states that “[t]he plant options are not limited to those above.” (FORTUM 2008, 40).

This is very vague because reactor type and output are not determined. It is not possible to properly assess transboundary impacts with precise information on power output and reactor type missing.

The Finnish procedure allows the applicant to deal with the different reactor projects as a “black box” which has to follow Finnish nuclear regulations. This is convenient for the applicant because the responsibility for meeting the safety targets lies with the suppliers, in particular if Fortum can negotiate a turn-key contract for LO 3. The Finnish regulations are regarded as very strict but, beside limited emission targets, there seem to be no detailed safety requirements published. For orientation, the Finnish authorities should provide a comprehensive list of the specific safety requirements for Generation III reactors in Finland.

According to the Finnish EIA act, the EIA Directive of the EC and the ESPOO Convention, information about a planned project with potential transboundary impacts has to be given for assessing these impacts. If information about technical and safety properties of the reactor types in discussion is left out, this information cannot be considered complete. Therefore, it might be questionable whether this type of description of the project complies with the legal demands.

According to EC 2003/4 all environmental information should be made available to the public. Therefore, the missing information could also be claimed with reference to the Aarhus Convention and the corresponding EC Directive 2003/4, respectively.

It might be questionable if a transboundary EIA can be conducted under these circumstances.

Reactor Types for LO 3

The EIA Report presents only general information about the project. The NPP is presented as a black box, which has to meet the Finnish regulations and requirements. Without any description of the NPP's features it is not possible to assess the feasibility of realization of this target.

Furthermore, the EIA Report does not follow the recommendation of the Ministry of Trade and Industry: "In the Ministry’s view, the EIA Report should include a review of current nuclear power plants on the market which are suitable for the project under review. Similarly, the safety planning criteria for the prospective plant must be presented with respect to the limitation of emissions of radioactive substances and environmental impacts, as well as an assessment of the possibilities of meeting the safety requirements in force." (MTI 2007, 210).

In contrast to the Finnish procedure, a similar consultation process in the UK provides feasibility studies of Generation III reactors at a public website[2]. The British authorities have made comprehensive documents about the reactor types in discussion available to the public.

In the Finnish procedure feasibility studies for the reactors have to be provided by the applicant (Pöllänen 2008). It is recommended to make the information available for the public and for the ESPOO partners as soon as possible.

The EIA Report does not provide sufficient information about the reactors considered for LO 3. Based on the information given in the EIA, an evaluation of safety, the maximal source term and its probability of occurrence is not possible. Chapter 4 of this Expert Statement provides basic information on these reactors, researched by the authors using public literature.

There is very little operational experience with the reactor types listed in the EIA. Only one of those types (ABWR) has been in operation so far.

The majority of the reactor types listed relies on ex-vessel cooling (i.e., installation of a core catcher) for the control and mitigation of severe accidents. Fundamental problems still remain regarding the reliable functioning of a core catcher. There are no recent in-depth discussions on these topics. This could imply that no substantial progress has been achieved in solving the problems. The concept of in-vessel cooling is basically more promising, but difficult to implement in large reactors.

Moreover chapter 4 of this statement presents an extended discussion concerning special features, PSA results and potential problems of two reactors: the EPR and the ESBWR.

These two reactors have been chosen not only because they are of different type, PWR and BWR, respectively, but because they have fundamentally different safety concepts: The EPR is the most conservative of the proposed reactors. This reactor is an evolutionary development of the German KONVOI and the French N4 reactor and relies mainly on active safety systems. The ESBWR is among the most advanced new designs and relies mainly on passive safety systems.

The **EPR** has, compared to earlier PWRs, one significant new feature: The core catcher, for ex-vessel corium cooling. But the EPR's core melt frequency lies in the same range as the CDF reported, for example, for newer German PWRs. This implies that another new feature of the EPR, the in-containment refuelling water storage tank, does not have a significant effect on overall plant safety. According to PSA results available, the reduction of the large release frequency due to the core catcher, again compared to newer German PWRs, is less than a factor of two. The main advantage of the measures for core melt mitigation at the EPR seems to be that filtered venting can be avoided.

The **ESBWR** has been developed from another Generation III BWR (the ABWR) and is equipped with numerous new design features. It relies, to a considerable extent, on passive safety systems. ESBWR does not require operator action for successful mitigation until 72 hours after the onset of an accident. The ESBWR is also equipped with a core catcher with a simpler design than that of the EPR. The core damage frequency of the ESBWR, as reported, is considerably lower than CDFs of Generation II reactors. The large release frequency appears to be well

[2] http://www.hse.gov.uk/newreactors/reactordesigns.htm, seen 05-08-2008

below 10E-8/yr, and hence is also considerably lower than that of Generation II reactors. The flipside of the ESBWR's apparently high safety is that this reactor type is still in an early planning stage.

Safety and Accident Analysis

PSA results for the reactor types can be found in the open literature. Core damage frequencies span two orders of magnitude. The reactor types at the upper limit (EPR, APR-1400) lie in the same range as reported for newer Light Water Reactors of Generation II.

The source term, as assumed in the EIA Report for exemplary dose calculations, appears questionable regarding the relation between Cs-137 and I-131; and it does not take all nuclides into account, which are required for checking European Utilities Requirements (EUR) release criteria (Criteria for Limited Impact). Even so, not all EUR criteria are kept by the source term in the EIA Report.

Regarding the impact of a commercial airliner on an EPR, the authority reported that numerous details still need to be finalized, analyses need to be completed, and the results of the analyses must be verified experimentally (STUK 2005).

It is beyond the scope of this expertise to discuss whether it can be expected that the reactor types listed in the EIA Report conform to Finnish regulations. A detailed technical assessment of the reactors would be required to answer this question.

In the EIA Report a dose assessment of the accidental release with the exemplary source term is presented for a region of 100 km. For a distance up to 1,000 km the dose is evaluated by extrapolation and estimations. This assessment is not state of the art. The information of the EIA Report is not sufficient for an assessment of the potential impact of a severe accident at the LO 3 NPP. A worst case has to be discussed and the source term has to be proved to be the maximum release. Sufficient information about the used weather conditions and the type of dispersion model used by the computer programs have to be provided. For modelling the long-range transport, diffusion and deposition of radionuclides more sophisticated tools are required suitable for meso-scale to global-scale calculations. The dose values shall be published together with their calculated errors. However, it is unclear whether one of the three models used for the assessment is appropriate for modelling long-range transport and dispersion of radionuclides.

Moreover, the Ministry of Agriculture and Forestry suggests an assessment of threats posed by changes and the rise in the sea water levels because these have an impact on the reliability and safety of the nuclear power plant (MTI 2007). The present EIA for LO 3 does not sufficiently cover the consequences of a possible rise in the sea water level due to climate change. This suggestion of the Ministry of Agriculture and Forestry is very important. We support the demand for a comprehensive assessment of threats posed by changes and a rise in the sea water levels.

Last but not least, we suggest taking into consideration a description of how Austria, among other countries, will be informed in case of emergency.

Nuclear fuel chain

The EIA Report states that the uranium resources will be sufficient for several hundred years.

But uranium reserves are limited. An increase of uranium production of at least some 50% would be required in order to match only the future demand of current nuclear capacity (EWG 2006).

Thus the low availability of uranium could be an obstacle for continuing or expanding the use of nuclear power. In addition to that, new uranium production will require a much higher price level, and is also a matter of proliferation risk.

Uranium mining and nuclear fuel production activities have an immense impact on the land where it is mined and on people living there, irregardless of the mining method. Moreover uranium mining and fuel production need energy and thus emit CO_2. CO_2 emissions depend on the quality of ore and, since high grade ore is scarce, the CO_2-balance of uranium is becoming worse.

Construction of a final repository is intended at Olkiluoto. At present Finland is investigating the bedrock at Olkiluoto in order to locate the repository there. This is assumed to be ready for disposal in 2020. There is no discussion in the EIA Report of alternatives should the bedrock be found inadequate for long-term safety.

Meanwhile, the spent fuel is stored in interim storage at the Loviisa nuclear power plant site. Storage in a pool is not an optimal technology for long-term interim storage. Critical aspects are the integrity of the fuel rods and their handling after several decades in the pool. According to the EIA Report, a further extension of the interim fuel storage is envisaged in order to prepare a place for the fuel from LO 3. Since it is planned to store the spent fuel in interim storage over several decades up to 60 years, the disadvantage of the storage pool compared to a dry one should be considered. Furthermore, an assessment of the risk of accidents caused by external impacts to the pool storage should be given.

The EIA Report merely states that the long-term safety of the final repository has been proven by using a model for the calculations.

The EIA Report states that the long-term safety of the final repository has been assessed by making a very conservative safety analysis and justification. Anyway, there have to be large uncertainties in the results of the safety analysis. The assessment of the influence of ice-ages with fault movements, land uplift, earthquakes and the creation of new weakness zones also contributes to these uncertainties. Unfortunately the uncertainties are neither provided nor discussed.

In particular, the long-term capability and behaviour of the technical and geological barriers cannot be guaranteed because of the long storage period required. A discussion of uncertainties should be implemented within the EIA.

Alternatives and zero option

The zero option and alternatives are discussed in the EIA Report: Alternatives regard the reactor type (boiling or pressurized reactors) and the bandwidth of 1,000 MW to 1,800 MWe.

As alternatives, the EIA Report regards mainly the alternative locations of the outlet and intake of cooling water.

No alternative options for electricity production or options for investment in energy efficiency are discussed in the EIA Report. It was explained that this was in the competence of the Ministry of Employment and the Economy (MEE).

During the consultation of Austria and Finland in May 2008 a presentation of STUK addresses the Finnish energy policy (PÖLLÄNEN 2008): The new Finnish Climate and Energy Strategy is not ready by now, it will be dealt by the Parliament in autumn, 2008. The decision-in-principle on new nuclear power plants probably depends on the decision on the energy strategy.

Regarding Fortum´s statement that nuclear energy is CO_2 free, it has to be stated that if the total nuclear chain and the management of all types of radioactive waste are considered, nuclear energy cannot be regarded as CO_2 free and environmentally sound. In particular uranium mining and fuel fabrication cause a significant impact on the environment (see also chapter 6 of this statement). A complete comparison of the real costs and risks of nuclear energy production with those of renewable energy production alternatives proves that there is no advantage of nuclear electricity over renewable energy (BMLFUW 2007).

Fortum uses uranium from Russia for the Loviisa plant and therefore creates CO_2 emissions of 65 g/kWh (FRITSCHE 2007).

Nuclear energy also proves to be a comparatively costly measure to reduce CO_2 emissions. Energy efficiency measures, renewable energies and alternative solutions in the wider sense replace 2.5 to 10 times as much CO_2 per unit investment (BMLFUW 2007).

3 THE PROCEDURE

3.1 Treatment in the EIA Report

According to the Finnish law the construction of a new nuclear power plant is subject to a decision-in-principle[3] issued by the Government and ratified by the Parliament. The EIA process has to be completed before the decision-in-principle concerning a new nuclear power plant can be issued.

The first stage of the EIA process (assessment programme) was completed with the issuing of the Statement of the Ministry of Trade and Industry[4] in October 2007. This Statement included the summarized comments of all organizations on the EIA programme.

The second stage of the EIA procedure started with the preparation of the EIA Report which was submitted in April 2008. This part of the procedure including the ESPOO procedure is still under way. It will be concluded with another Statement of the Ministry of Employment and the Economy.

The prospective new unit at Loviisa will be a light water reactor; either a pressurized water reactor (PWR) or a boiling water reactor (BWR) (FORTUM 2008, 37f.). A service life of 60 years is envisaged, the electric power is planned to be approx. 1,000 to 1,800 MW (FORTUM 2008, 41).

In chapter 3 of the EIA Report ten different plant types are listed. Fortum states that "[t]he plant options are not limited to those above." (FORTUM 2008, 40).

All the reactors listed are generally considered to belong to Generation III. There is no review of these reactors and no information about technical and safety features in the EIA Report.

The new NPP can be constructed for combined heat and power production (FORTUM 2008, 5). But it is an open question whether there is a demand of heat near the NPP.

3.2 Discussion

Not only Fortum is planning a new NPP, but also Teollisuuden Voima Oy (TVO) and Fennovoima Oy have started EIA procedures for new NPPs. TVO is planning a fourth unit at Olkiluoto (see also WENISCH et al. 2008b), Fennovoima Oy proposed four different sites in Finland for a new NPP (WENISCH & KROMP 2008b).

Both Fortum and TVO argue that the Finnish electricity need will increase. Fortum states that electricity consumption in Finland in 2007 was 90.3 TWh with estimated growth to 115 TWh in 2030 (FORTUM 2008, 4). TVO estimated to exceed 100 TWh in 6 to 8 years (TVO 2008, 9).

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[3] In the EIA Report it is stated that a decision-in-principle is now called favourable resolution. In the Nuclear Energy Act it is still named decision-in-principle.

[4] since 2008: Ministry of Employment and the Economy

Fortum also wishes to construct the new NPP to replace the older NPPs in Loviisa and also to replace fossil-fuel fired plants (FORTUM 2008, 20).

A NPP of 1,800 MW would produce at best 14 TWh electric energy per year. The two existing reactors at Loviisa generate about 8 TWh per year. It is the responsibility of the Finnish Ministry of Economy and Employment, the Government and the Parliament to decide when and what capacity will be required to serve the electricity demand, and how many NPPs shall be built and where.

Because these three EIA procedures all deal with the same topic – the planning of new NPPs to meet Finland's increasing electricity need – they should be dealt with together. This is also the position of the Ministry of Employment and the Economy. The Minister of Economic Affairs, Mauri Pekkarinen, would have preferred the completion of the EIA process for Olkiluoto-4 before receiving the application by TVO for the decision-in-principle, as he told in a press release of the Ministry of Employment and Economy of April 25th 2008[5]. Concerning the other two expected applications (LO 3 NPP by Fennovoima Oy) for decisions-in-principle the Minister announced that it was "essential for all applications to be considered at the same time". So even if TVO submitted earlier than the other operators, a joint consideration will take place.

In chapter 6 the planned NPP by Fennovoima Oy is mentioned because one of the possible construction sites is near Loviisa. "With the implementation plans and decisions lacking, no combined effects with Fortum´s LO 3 project have been assessed in this report. Other projects having potential combined effects are not known."

Fortum does not present a detailed list of reactor types and their technical specifications, only a non binding list of ten plant types. Also the Ministry of Trade and Industry declared in its statement for the EIA Assessment Program: "In the Ministry's view, the EIA Report should include a review of current nuclear power plants on the market which are suitable for the project under review. Similarly, the safety planning criteria for the prospective plant must be presented with respect to the limitation of emissions of radioactive substances and environmental impacts, as well as an assessment of the possibilities of meeting the safety requirements in force." (MTI 2007, 210)

The Finnish procedure allows the applicant to deal with the different reactor projects as a "black box" which has to follow Finnish nuclear regulations. This is convenient for the applicant because the responsibility for meeting the safety targets lies with the suppliers, in particular if Fortum can negotiate a turn-key contract for LO 3. The Finnish regulations are regarded as very strict but beside limited emission targets there seem to be no detailed safety requirements published. For orientation the Finnish authorities should provide a comprehensive list of the specific safety requirements for Generation III reactors in Finland.

Since there is no information about the reactors under review for LO 3, chapter 4 of this statement provides basic information on this issue and an exemplary more detailed discussion of a Generation III PWR (EPR) and BWR (ESBWR). These two reactors have been chosen because they have fundamentally different safety concepts: the EPR is the most conservative of the proposed reactors; it is an evolutionary development of the German KONVOI reactor and the French N4 reactor

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[5] http://www.tem.fi/?89521_m=91497&l=en&s=2471, seen 05-09-2008

and relies mainly on active safety systems. The ESBWR has the most advanced new design and relies mainly on passive safety systems. The comparison of these two reactors shows the bandwidth of advantages and disadvantages of the new features of the proposed reactors.

In contrast to the Finnish procedure a similar consultation process in the UK provides feasibility studies of Generation III reactors at a public website[6]. The British authorities have made comprehensive documents about the reactor types in discussion available to the public.

In the Finnish procedure feasibility studies for the reactors have to be provided by the applicant (Pöllänen 2008). It is recommended to make the information available for the public and for the ESPOO partners as soon as possible.

More information about the reactor is important because the nuclear inventory is different for a reactor with an output of 1,000 or 1,800 MW. The inventory is of importance for the assessment of potential transboundary impacts also on Austria.

The EIA procedure should be in accordance with the Finnish EIA Act, the ESPOO convention and the Aarhus Regulation.

The *Finnish EIA Act* (EIA Act 2006) corresponds to the EIA Directive of the EC (EC 97/11) by including nuclear power stations into the list of projects that are subject to the EIA legislation. The notification of Member States that are possibly affected by the project has to include information on the project and on any transboundary environmental impact, information on the assessment procedure and the time period for commenting this information.

The *ESPOO Convention* (ESPOO Convention 1997) lists activities that should be subject to a transboundary EIA process. These activities include nuclear power stations. The ESPOO Convention states in Article 6 (Final Decision): “The Parties shall ensure that, in the final decision on the proposed activity, due account is taken of the outcome of the environmental impact assessment.”

The *Aarhus Convention* guarantees the right of environmental information for the public. The Aarhus Convention is an international legal Convention. The United Nations Economic Commission for Europe (UNECE) “Convention on Access to Information, Public Participation in Decision-Making and Access to Justice in Environmental Matters” was adopted on 25 June 1998 in the Danish city of Aarhus and entered into force on 30 October 2001. At the end of 2006 40 parties have signed the Convention, including all Member States of the European Union (HÖRMAYER et al. 2007).

In the Aarhus Regulation (EC 1367/2006) all three pillars of the Aarhus Convention are covered: access to information, public participation, access to justice. In 2003 the European Commission adopted two Directives concerning the first and second pillars of the Aarhus Convention, the right of access to environmental information and the right of public participation. The Directives are implemented into national law of the EU Member States (EC 2003/4 und EC 2003/35[7]).

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[6] http://www.hse.gov.uk/newreactors/reactordesigns.htm, seen 05-08-2008

[7] EC Directive 2003/35 deals with the right of public participation in respect of the drawing up of certain plans and programmes relating to the environment (EC 2003/35).

EC Directive 2003/4 on public access to environmental information (EC 2003/4) grants the right of access to environmental information. Environmental information includes according to Article 2 any information on “factors, such as substances, energy, noise, radiation or waste, including radioactive waste, emissions, discharges and other releases into the environment, affecting or likely to affect the elements of the environment”. If a Member State disobeys central elements of the Directives 2003/4/EC and 2003/35/EC, any natural or legal person will be allowed to lodge a complaint with the Commission against this Member State (HÖRMAYER et al. 2007).

3.3 Conclusion

It is questionable if the project is elaborated enough to fulfil the requirements of the EC EIA Directive (EC 97/11) and the ESPOO Convention (ESPOO Convention 1997), because the EIA Report does not present a certain project and alternatives. In the EIA Report only a “non binding” list of ten reactor types is presented, with an approximate output of 1,000–1,800 MWe. This is very vague because reactor type and output are not determined. It is not possible to properly assess transboundary impacts with this missing information.

The EIA Report presents only general information about the project. The NPP is presented as a black box, which has to meet the Finnish regulations. The Finnish regulations are regarded as very strict but beside limited emission targets there seem to be no detailed safety requirements published. For orientation the Finnish authorities should provide a comprehensive list of the specific safety requirements for Generation III reactors in Finland.

Without any description of the NPP's features it is not possible to assess the feasibility of realization of this target. It is questionable whether this type of description of the project complies with the EC EIA regulation. Furthermore, the EIA Report does not follow the recommendation of the Ministry of Trade and Industry.

The Ministry for Employment and the Economy wants to consider all three applications for a decision-in-principle together (OL 4, LO 3, NPP by Fennovoima Oy). We encourage this approach of the Ministry.

According to the Finnish EIA act, the EIA Directive of the EC and the ESPOO Convention information about a planned project with potential transboundary impacts has to be given for assessing these impacts. If information about technical and safety properties of the reactor types in discussion are left out, this information cannot be considered as complete. Therefore, it is questionable whether this type of description of the project complies with the legal demands.

According to EC 2003/4 all environmental information should be made available to the public. Therefore the missing information could also be claimed with reference to the Aarhus Convention and the corresponding EC Directive 2003/4, respectively.

It might be questionable if a transboundary EIA can be conducted under these circumstances.

4 REACTOR TYPES FOR LO 3

4.1 Treatment in the EIA Report

Chapter 3.2.3 Commercial Plant Options of the EIA Report (FORTUM 2008) lists some plant types on the market. This listing is not binding, however, and other suppliers may also come into question. All the reactors listed are generally considered to belong to the Generation III (almost all commercial nuclear power reactors under operation today belong to the Generation II).

Information on the reactor types mentioned in the EIA Report is presented in the following tables. This information was researched by the authors and is taken mostly from publications of plant designers and other nuclear industry sources and from IAEA.

A short discussion of the main features of the reactor types follows. Within the scope of this Expert Statement, this discussion has to remain limited. After that, an extended discussion can be found for the EPR and the ESBWR. Among others, the following issues are discussed in more detail

- Special features
- PSA results
- Indications for potential problems.

These two reactors have been chosen not only because they are of different type, PWR and BWR, respectively, but because they have fundamentally different safety concepts: The EPR is the most conservative of the proposed reactors, It is an evolutionary development of the German KONVOI reactor and the French N4 reactor and relies mainly on active safety systems. The ESBWR has the most advanced new design and relies mainly on passive safety systems. The comparison of these two reactors shows the bandwidth of advantages and disadvantages of the new features of the proposed reactors.

Notes to the tables

NRC certification: Information regarding NRC certification has been taken from a paper of the World Nuclear Association (WNA 2008).

Units existing: This includes units in operation, under construction or firmly planned with start-up of construction in near future.

Special features: The most important features which go beyond Generation II plants are listed.

Table 1: European Pressurized Water Reactor.

EPR	**European Pressurized Water Reactor (Evolutionary PWR)**
Basic data	PWR, ca. 1,700 MWe
Manufacturer	AREVA (France/Germany)
Origin	Developed from the German KONVOI and French N4 PWR types
Certification	EUR certified NRC certification process ongoing (WNA 2008)
Units existing	2 units under construction: Olkiluoto 3 (Finland); start of construction 2005, original estimate of start-up 2009. Due to problems with quality control in 2006 and further delays in 2007, the schedule slipped to about 2011 so far (NEIMAG 2007). Flamanville (France); start of construction 2007, expected start-up 2012 (WNIH 2008). Schedule threatens to slip due to problems with quality control similar to those at OL-3.
Special features	Core-catcher for reactor core in case of meltdown In-containment refuelling water storage tank (combines coolant storage and sump function – switchover from safety injection to sump recirculation is avoided) Double containment (two concrete hulls) (EDF 2006)
PSA results	Olkiluoto 3 CDF (external and internal initiators, operation and outages) = 1.8E-06/yr Frequency of exceeding release limit (100 TBq Cs-137, plus other nuclides) = 1.0E-07/yr (STUK 2005) Flamanville CDF (ext. and int. initiators, op. and out.; seismic analysis not complete, internal explosions not included) = 1.33E-06/yr (EDF 2006) The same value is given for the EPR applied for in the UK (UK-EPR 2008)

Table 2: Vodo-Vodyanoy Energeticheskiy Reactor.

VVER-1000/392M (AES-2006)	**VVER = Vodo-Vodyanoy Energeticheskiy Reactor**
Basic data	PWR, ca. 1,150 MWe
Manufacturer	Gidropress/Atomenergoproekt (Russia)
Origin	AES-2006 was developed from the AES-92; the AES-92 was developed from the standard VVER-1000/320
Certification	EUR certified
Units existing	Two units under construction (contract signed June 2007): Novovoronezh-2 units 1 and 2 (commercial operation planned 2012/2013) (In the World Nuclear Industry Handbook, the first unit is listed as “under construction”, the second one as “reasonably firmly planned” (WNIH 2008).)
Special features	Combination of passive and active safety mechanisms (e. g., passive SG heat removal, passive core cooling systems) Core-catcher for reactor core in case of meltdown (“melt retention in a special device located beneath the reactor vessel”) Double containment (two concrete hulls)
PSA results	“(G)eneral frequency of core damage at a level of 1.0E-07 (/yr).” All categories of initiating events, power and shutdown: CDF = 5.4E-08/yr (IAEA 2004)

Reference: (GENERALOV 2007) unless specified otherwise

Table 3: Advanced Passive reactor.

AP-1000	**AP = Advanced Passive**
Basic data	PWR, 1,100 MWe
Manufacturer	Westinghouse (USA)
Origin	Developed from AP-600. More innovative than other reactor types discussed here; not directly developed from a Generation II plant
Certification	EUR certified NRC design certification December 2005
Units existing	No units in operation or under construction yet. 4 units “firmly planned” in China. Sanmen-1, -2: Start of construction 2009, commercial operation 2013; Haiyang-1, -2: No data given (WNIH 2008)
Special features	Relies on passive safety systems to a large extent – e. g. passive core cooling, containment isolation, containment cooling system, MCR emergency habitat system. Most, but not all valves aligning the safety systems are fail-safe “Simplified design” (50% fewer valves, 35% fewer pumps, 80% less pipes, 45% less building volume, 70% less cable) Increased safety margins in case of DBAs In-vessel retention of damaged core external cooling of RPV with inventory from in-containment refuelling water storage tank (BRUSCHI 2004, WEC 2007)
PSA results	CDF = 5.0E-07/yr, LRF = 6.0E-08/yr (WEC 2007) CDF = 4.0E-07/yr (BRUSCHI 2004) LRF = 1.95E-08/yr (IAEA 2004) (in all three cases, no specification regarding inclusion of external/internal, operation/shutdown are provided)

Table 4: Advanced Boiling Water Reactor.

ABWR	**Advanced Boiling Water Reactor**
Basic data	BWR, 1,400–1,600 MWe
Manufacturer	Hitachi/Toshiba/General Electric (Japan/USA)
Origin	Originally designed by GE, developed from older GE BWR designs
Certification	EUR certified NRC certified
Units existing	5 in operation, all in Japan (begin of commercial operation): Kashiwazaki-Kariwa-6 (1996), -7 (1997) Hamaoka-5 (2005) Higashidori-1 (2005) Shika-2 (2006) 4 under construction (2 in Japan, 2 in Taiwan): Fukushima-Daiichi-7, J (start-up planned 2006?) Shimane-3 (2011), J Lungmen-1, -2, T (2009, 2010) 2 “firmly planned” in Japan: Kaminoseki-1, -2 (start of construction 2009/2012, operation 2014/2017) (WNIH 2008)

ABWR	Advanced Boiling Water Reactor
Special features	"Simplified active safety systems". In case of LOCA, plant response has been fully automated and operator action is not required for 72 hours, the same capability as for passive plants (DNE 2008) Some passive severe accident mitigation features (BEARD 2007) Spreading area in lower drywell and passive drywell flooding system to guarantee coolability of core debris (IAEA 2004, BEARD 2007) This feature seems to apply to the US ABWR only. The sources above are not fully clear in this respect, but a paper on Kashiwazaki-Kariwa does not mention a capability of ex-vessel core cooling (TSUJI 1998).
PSA results	Internal events CDF = 1.6E-07/yr, high seismic margins claimed, LRF < 1.0E-9/yr (The contribution of mode 6 (refuelling) to CDF is reported to be 99%, so no level 2 (PSA) would be required.) (BEARD 2007)

Table 5: Siedewasserreaktor.

SWR-1000	SWR = Siedewasserreaktor
Basic data	BWR, ca. 1,000 MWe
Manufacturer	AREVA (Germany)
Origin	Developed by Siemens-KWU in the 1990s, based on the concept of the SWR-300 The SWR-300 was developed in the 1980s as a small, inherently safe BWR
Certification	EUR certified
Units existing	No units are in operation, under construction or firmly planned today
Special features	Passive safety systems – e.g. containment cooler, passive flooding and emergency condensers for core cooling, passive pulse generator for initiation of safety systems (The reactor, however, does not entirely rely on passive systems for accident control; there is a combination of active and passive measures. It is claimed that passive systems and active systems each are alone sufficient to provide adequate cooling of the reactor core in case of an accident.) In-vessel retention of damaged core – external cooling of RPV by flooding of the reactor shaft (passive via the containment cooler) (BRETTSCHUH 2001)
PSA results	CDF for internal events = 1.1E-07/yr (5.0E-08/yr for power operation, 6.0E-08/yr for shut-down) (BRETTSCHUH 2000, 2001)

Table 6: Economic Simplified Boiling Water Reactor.

ESBWR	**Economic Simplified Boiling Water Reactor**
Basic data	BWR, ca. 1,600 MWe
Manufacturer	General Electric (USA)
Origin	Developed from GE SBWR (Simplified BWR) and ABWR (see above)
Certification	NRC certification process ongoing, design certification expected 2009 or 2010 (WNA 2008)
Units existing	No units are in operation, under construction or firmly planned today
Special features	Passive safety systems (e. g. passive core cooling with GDCS (gravity-driven cooling system); passive containment cooling system No operator action needed for design basis accidents for 72 hours (IAEA 2004) Core catcher with passive flooder Generally – reduced and simpler systems, reduced materials and buildings
PSA results	CDF = 3.0E-08/yr (no specification internal/external or plant state) CDF = 6.16E-08/yr (without external hazards; 45% at-power, 55% shutdown; 34% internal events, 66% internal hazards) (UK-ESBWR 2008)

Reference: (HINDS 2006) unless specified otherwise

Table 7: Advanced Pressurized Water Reactor.

APWR	**Advanced Pressurized Water Reactor**
Basic data	PWR, 1,600–1,700 MWe
Manufacturer	Mitsubishi
Origin	Developed from Mitsubishi PWRs (Westinghouse was involved earlier) For the US market, MHI developed the US-APWR, a slightly modified APWR complying with US regulations
Certification	EUR certification process ongoing (submitted for design certification March 2008 (AUA 2008)) NRC certification process ongoing
Units existing	Two units are definitely planned in Japan (Tsuruga-3 and -4, start of construction reported as 2007, start of operation 2014 and 2015) (WNHI 2008) License application for the first two US-APWRs (site in Texas) expected for 2008 (NEI 2007)
Special features	Simplified ECCS – integrating low pressure injection systems and accumulators In-containment refuelling water storage tank (combines coolant storage and sump function – switchover from safety injection to sump recirculation is avoided) Floor below reactor cavity with 1 m thick protective layer of concrete for molten debris; to be cooled there from the fire service water system. Molten debris will be coolable; erosion of concrete can be prevented. Outlet from RPV cavity to containment considered to be constructed like a labyrinth. (IAEA 2004)
PSA results	CDF expected to be at least one order of magnitude lower than for existing 4-loop PWRs, i.e. about 1.0E-07/yr (IAEA 2004)

Table 8: Advanced Power Reactor.

APR-1400	**APR = Advanced Power Reactor**
Basic data	PWR, 1,400 MWe
Manufacturer	KHNP (South Korea)
Origin	Developed from the type PWR 80+, which was developed by the US firm ABB
Certification	No EUR or NRC certification PWR 80+ has been certified by NRC (OLSON 1997)
Units existing	Two APR-1400 are firmly planned in South Korea (Shin-Kori 1 and 2, start of construction reported as 2006/2007, operation 2010/2011) (WNIH 2007, 2008)
Special features	External reactor vessel cooling system (ERVCS) for in-vessel retention of corium; plus back-up system (CFS) for flooding corium in the cavity below reactor vessel, if ERVCS fails. CFS is gravity-driven. In-containment refuelling water storage tank (combines coolant storage and sump function – switchover from safety injection to sump recirculation is avoided) Safety systems appear to be mostly active (IAEA 2004)
PSA results	CDF for internal events = 2.25E-06/yr CDF for external events (bounding site characteristics) = 4,36E-07/yr Containment failure frequency from all events: 2.84E-07/yr (possibly for operation only; not clear if shut-down state included) (NEA 2002)

4.2 Discussion

4.2.1 Reactor Types

There is very little experience with the reactor types listed in the EIA Report; some of them so far exist only on paper.

Table 9: Overview of the status of realization of the reactor types considered in this expertise.

Status	**No. of types**	
In operation	1	ABWR
Under construction	2	EPR, AES-2006
Firmly planned	3	AP-1000, APWR, APR-1400
None of the above	2	SWR-1000, ESBWR

4.2.1.1 Core Catcher

There is still little experience in the realization of a “core catcher” for ex-vessel cooling of a molten core. It appears that the ABWRs taken into operation so far do not have this feature and that only future plants of this type will be equipped with it.

Fundamental problems regarding the functioning of a core catcher have been reported in the last years. They include (SEHGAL 2004, SEVON 2005):

- Interaction between molten core and concrete cannot be accurately simulated.
- There are high uncertainties regarding heat transfer rates.
- Cracking of the concrete surface can occur; this has not been studied systematically so far.
- Complications due to H_2 generation are possible.
- Cooling with water flooding alone (as generally planned) might be insufficient, cooling coils could be required.
- Violent steam explosions can take place before the core reaches the catcher.

Within the scope of the internet research the authors performed, no more recent in-depth discussions on these topics could be found.

The concept of in-vessel cooling, i.e. external cooling of the reactor pressure vessel in case of a severe accident to prevent discharge of the molten core basically seems to be more promising and not beset with so many problems. In-vessel cooling has already been implemented as severe accident management measure at the Loviisa NPP (2 units with about 500 MWe each) in Finland (CSNI 2002).

However, in-vessel cooling is difficult to implement in larger reactors, due to the surface-to-volume ratio getting less favourable with increasing power. In fact, for the reactor types considered here, there is a tendency of in-vessel cooling being planned for the smaller ones, ex-vessel cooling for the larger, with the AES-2006 being an exception to this rule.

Table 10: Overview of the molten core cooling strategies for each type.

Reactor type	Power (MWe)	
EPR	1,700	ex-vessel cooling
AES-2006	1,150	ex-vessel cooling
AP-1000	1,100	in-vessel cooling
ABWR	1,400–1,600	ex-vessel cooling
SWR-1000	1,000	in-vessel cooling
ESBWR	1,600	ex-vessel cooling
APWR	1,600–1,700	ex-vessel cooling
APR-1400	1,400	in-vessel and ex-vessel cooling

4.2.1.2 Core Damage Frequency

PSA results are reported for all reactor types listed in the EIA Report. Within the scope of the present expertise, it was not possible to research this area in detail. Hence, the results which could be found in the time available are not altogether comparable.

Table 11: Overview of core damage frequency and large release frequency as reported in different sources.

Reactor type	CDF per reactor year	LRF per reactor year
EPR (1)	1.33E-06–1.8E-06	1.0E-7
AES-2006 (1)	5.4E-08–1.0E-07	–
AP-1000 (4)	4.0E-07–5.0E-07	1.95E-08–6.0E-08
ABWR (3)	1.6E-07	<1.0E-09
SWR-1000 (3)	1.1E-07	–
ESBWR (3)	6.16E-08	–
APWR (4)	1.0E-07	–
APR-1400 (2)	2.69E-06	2.84E-07 (containment failure)

(1) internal and external initiators, operational and shutdown states

(2) internal and external initiators; not clear if shutdown states included

(3) internal initiators, operational and shutdown states

(4) no specification regarding events and states

Core damage frequencies as reported span almost two orders of magnitude, with the APR-1400's CDF of about 2.7E-06/yr as the highest– and the ESBWR's CDF of 6.16E-08/yr as the lowest.

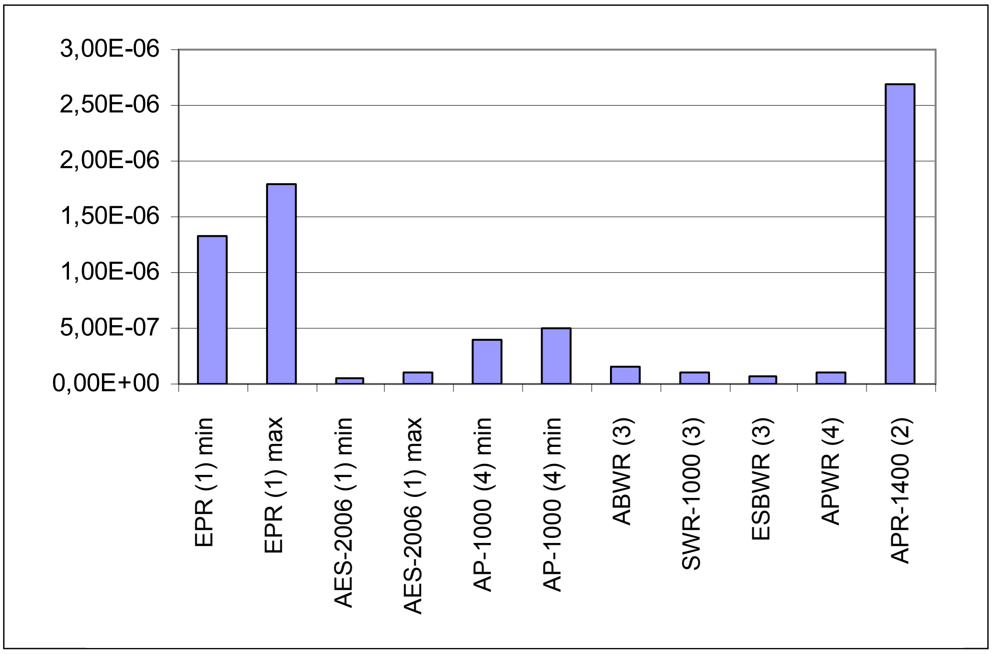

Figure 1: Core damage frequency of different Generation III reactors.

The CDF of the EPR is the second highest, only slightly below that of the APR-1400.

It is interesting to note that the CDF of APR-1400 and EPR lies in the same range as the CDF reported for operating light water reactors from Generation II. For example, CDF for the newer German PWRs and BWRs is about 2.0E-06–2.7E-06 per reactor year (CNS 2002).

When comparing the numbers, it has to be taken into account that in some cases, it is not clear how complete the PSA leading to the reported result has been. Regarding the ESBWR it has to be taken into account that no units of this reactor type are in operation, under construction or firmly planned so far. Due to this comparatively early planning stage, PSA results might be less reliable than results for further developed types.

On the other hand, a rather low number (about one order of magnitude below the EPR's) is also reported for the ABWR which is the only reactor type among those listed in the EIA Report already in operation. Within the scope of this expertise, it is not possible to obtain more detailed information on the ABWR's PSA results, and their trustworthiness.

4.2.2 Extended Discussion about EPR

4.2.2.1 Special features of EPR's

Compared to the predecessor types (French N4 and German KONVOI), the EPR has only a limited number of special features. The most important ones are the core catcher with the containment heat removal system, and the in-containment refuelling water storage tank (IRWST). Also, the construction of the containment differs from that of the predecessors.

Ex-vessel cooling of molten core (core catcher) and other measures for core melt mitigation

Following, a brief description of the ex-vessel cooling concept based on documents of the reactor designer is provided (UK-EPR 2008, 2.S.2.2.4).

To stabilize the molten core in a severe accident, the EPR relies on an ex-vessel strategy with a water-cooled core catcher. In-vessel melt retention by outside cooling of the RPV was dismissed because the high power rating of the reactor leads to low margins for heat transfer.

The molten material from the RPV is first collected in the reactor pit. In the pit, the corium is temporarily retained by a layer of sacrificial concrete. The time delay and the admixture of the concrete leads to a collection of melt in the pit and a more uniform spectrum of possible melt states at the end of the retention process. Finally, the melt will penetrate the melt plug consisting of concrete and a metal plate (of Al/Mg-alloy) and flow into the core catcher proper.

Because of the retention and collection in the pit, the subsequent spreading and the stabilization measures are largely independent of the uncertainties associated with in-vessel melt pool formation and RPV failure; there is a one-step release into to spreading area.

The core catcher is a shallow crucible, lateral to the pit. Its bottom and sidewalls are assembled from individual elements made of cast iron and covered with sacrificial concrete. There, the spread melt is to be stabilized by flooding and external cooling.

The total mass of metal and oxide melts is about 400 Mg (60–70 m³), spreading onto an area of 170 m².

The cooling of the melt in the core catcher by overflow of water from the in-containment refuelling water storage tank (IRWST) is fully passive and triggered by the arrival of melt in the core catcher. The water first fills the central supply duct underneath the core catcher, then enters the horizontal cooling channels and submerges the space behind the sidewalls. After filling, it will overflow onto the surface of the melt.

Alternatively to the IRWST, the containment heat removal system (CHRS) can be used to actively deliver cooling water.

Solidification of the melt is to be achieved within a few days (BITTERMANN 2003). After stabilization, long term cooling and heat removal follow.

The source quoted above describes analyses of all individual steps of the core cooling sequence. Results of experimental investigations are referred to. For several steps, computer codes were used for the validation strategy; manual calculations appear to have been employed in many cases. In some cases, several diverse approaches were applied.

Uncertainties are mentioned frequently, for example concerning the accident scenario and the initial state of the core; the sequence of melt release from the RPV; composition and state of the melt in the reactor pit; melt flow rates and melt physical properties in the spreading area. It is attempted to take them into account by making the melt stabilization concept tolerant of uncertainties, by considering bounding cases or by dealing with them in a probabilistic framework.

The ex-vessel core cooling system has to be seen in connection with the Containment Heat Removal System (CHRS). This system controls the containment pressure. It consists of a spray system and allows recirculation through the cooling structure of the molten core retention device to mitigate the consequences of the considered accident scenario (UK-EPR 2008, 1.F).

The CHRS serves to avoid containment failure while the molten core is stabilized in the core catcher. It also permits to avoid venting of the containment; KONVOI and N4 plants are equipped with filtered venting systems for containment pressure control.

Further measures for core melt mitigation to be implemented at the EPR are (UK-EPR 2008, 1.A):

- Prevention of high pressure core melt by high reliability of decay heat removal systems, complemented by primary system Overpressure Protection (OPP);
- Primary system discharge into the containment in the event of a total loss of secondary side cooling;
- Prevention of hydrogen detonation by reducing the hydrogen concentration in the containment at an early stage with catalytic hydrogen recombiners;
- Collection of leaks and prevention of bypass of the confinement, achieved by double-walled containment.

All of these features, however, are not particular to Generation III plants. They already are implemented in the predecessor types (KONVOI and N4).

In-containment refuelling water storage tank (IRWST) (UK-EPR 2008, 1.F & 2.B.2)

The IRWST pool is a reservoir containing a large quantity of borated water. It serves to collect water discharged into the containment in case of accidents (both design basis and severe). It is located so as to permit direct suction from the tank by the emergency core cooling system and other safety systems; thus, it combines the coolant storage and sump functions of the predecessor types.

Because of this combination, the requirement for switchover from safety injection to sump recirculation in case of an accident is avoided. The system design can be simplified; fewer valves are required, removing potential sources of failure.

The IRWST also supplies water for passive cooling of the molten corium in the core catcher.

Double containment (two concrete hulls)

The EPR containment is double-walled, founded on a basemat (foundation raft).

The inner containment is a pre-stressed concrete structure with a steel liner installed on the inner surface (including the basemat). Its function is to provide leak-tight-ness (leak rate of 0.3% per day) as well as resistance to internal pressure (design pressure 5.5 bar abs.). The volume is about 80.000 m^3.

The outer containment shell is a reinforced concrete structure. It has the function to provide protection against external hazards such as airplane crash and explosion pressure wave (UK-EPR 2008, 1.F).

The containments are separated by an annular space with a width of 1.80 m (UK-EPR 2008, 2.C.5.1).

Both containments have a wall thickness of 1.30 m. In chapter 1.A of UK-EPR 2008, the thickness of the outer containment is given as 1.80 m; this appears to be an error (mix-up with annulus width), since 1.30 m is stated in another source (AREVA 2005) for the outer containment.

Of the predecessor types, the N4 plants also have a double wall concrete containment. The inner containment is made of pre-stressed concrete without steel liner; the outer containment is a reinforced concrete structure (UK-EPR 2008, A).

Thickness of the inner containment is 1.20 m for the wall and 0.90 m for the dome; of the outer containment, 0.55 m for the wall and 0.40 m for the dome (IAEA 1992).

KONVOI plants have a double containment with a steel shell (thickness 38 mm) (BMU 2007) inside and a reinforced concrete wall (thickness 1.80–2.00 m) outside.

4.2.2.2 Probabilistic design targets and PSA results of EPR's

This section summarizes the most recent information published on probabilistic safety objectives and PSA results for the EPR (UK-EPR 2008).

Probabilistic design targets

Probabilistic design targets constitute guideline values for the verification and evaluation of the design; they must not be understood as strict design limits. The following probabilistic design targets are used for the EPR (UK-EPR 2008, 2.R.0):

Design targets linked to the risk of core damage:

- For internal events (i.e. not including internal and external hazards): An overall CDF of less than 1E-6/yr for power operation states, a CDF lower in shutdown states than in power operation states;
- Within this design target, the contribution of each family should not exceed 30%.
- For external hazards, an overall CDF of less than 5E-6/yr;
- For other events – particularly, internal hazards – an overall CDF of less than 3E-6/yr;

Design targets linked to the risk of loss of containment:

The core damage sequences are classified according to three generic consequences (PDS = Plant Damage State):

- PDS 1: core damage with mitigation features available (no loss of containment);
- PDS 2: low-pressure core damage with delayed loss of containment;
- PDS 3: core damage with potential early loss of containment.

In the scope of the level 1+ PSA (final results of the level 2 PSA are not available yet), it is specified that PDS 3 should be “practically eliminated”. A frequency target (solely for internal events) of 1E-7/yr is considered for PDS 3.

Design targets linked to the risk of large early releases:

For large, early releases without hazards, a frequency of less than 1E-7/yr,

for each accident situation, a balanced contribution to the risks of large early releases is aimed for; for this, a target value of a frequency of less than 1E-8/yr is used.

PSA results

Core damage frequency for internal events is 6.1E-7/yr

Main contributors are:

- LOCA: 23%
- ATWS: 20%
- Sec. system transients: 18%
- Loss of cooling systems, loss of ultimate heat sink: 16%
- LOOP: 14%
- Sec. side breaks: 4%
- Primary system transients: 4%
- Heterogeneous dilution: 1%

(UK-EPR 2008, 2.R.1)

Core damage frequency for internal hazards is 8.4E-8/yr (fire 76%, internal flooding 24%).

Core damage frequency for external hazards is 6.4E-7/yr (earthquake 78%, aircraft crash 10%, very cold weather, snow and wind 6%, sea ice 5%, others residual).

(UK-EPR 2008, 2.R.4)

Hence, probabilistic design targets for CDF are fulfilled, with an overall CDF of 1.33E-6/yr.

For plant damage states, the PSA level 1+ provided the following results (for internal events only):

- PDS 1 (no loss of containment): 85% of CDF (5.2E-7/yr)
- PDS 2 (delayed loss of containment): 9% of CDF (5.2E-8/yr)
- PDS 3 (potential early loss of containment): 6% of CDF (3.9E-8/yr)

(UK-EPR 2008, 2.R.2)

Probabilistic design targets for plant damage states are fulfilled.

Risk reduction categories

For the EPR, "plant condition categories" (PCC) correspond to what is usually designated as design basis. For PCCs, the safety analyses have to be performed with a conservative methodology (UK-EPR 2008, 2.P.0).

In order to reach the overall probabilistic targets, risk reduction categories (RRC) were introduced beyond the PCC. There are three RRCs:

RRC-A covers some multiple failure conditions not considered in the PCCs. Failure combinations and the unavailability of systems must be studied in a probabilistic manner to identify the sequences which can affect the overall probabilistic targets. If acceptance criteria are not met, additional design features have to be introduced. The analyses are to be performed with a realistic approach (UK-EPR 2008, 2.S.1.0).

RRC-B concerns accident situations likely to lead to large early releases. They have to be shown to be physically impossible; if this cannot be achieved, design measures must be taken to exclude them. The analyses are to be based on realistic calculations of physical phenomena (UK-EPR 2008, 2.S.2.0).

RRC-C covers events or sequences which are excluded from the design basis for deterministic or probabilistic reasons (for example, because of the implementation of the break preclusion assumption). They are analysed in order to introduce additional safety margins in the design.

The following events are considered:

- Double ended primary guillotine break (2A-LOCA)
- Double ended guillotine break of a main steam line (2A-MSLB)
- Three cases of containment bypass, involving steam generator tube rupture

(UK-EPR, 2.S.3.0)

The case of the 2A-LOCA is of particular interest (see below). In this case, analyses are carried out using best-estimate criteria and assumptions (UK-EPR, 2.S.3.2a).

4.2.2.3 Indications for potential weak points and problems of EPR's

Core damage frequency and large release frequency

It is interesting to note that the CDF of the EPR lies in the same range as the CDF reported for operating light water reactors from Generation II. For example, CDF for the newer German PWRs and BWRs is about 2.0E-6–2.7E-6 per reactor year (CNS 2002).

This implies that the introduction of the in-containment refuelling water storage tank (IRWST) as a measure to reduce complexity and increase safety does not have a significant effect on the CDF.

Thus, to the extent that the safety of the EPR is significantly improved compared to newer Generation II plants, this could only be with respect to better consequence mitigation capability in case of core melt.

According to PSA results for the EPR, there is no loss of containment for 85% of the internal events CDF. There is delayed loss of containment for 9% of internal events CDF, and potential early loss for 6%.

The corresponding results for a German KONVOI PWR (GKN II), with a comparable CDF, are as follows:

- Containment remains fully functioning: 33%
- Containment remains intact, filtered venting: 45%
- Unfiltered releases, late containment failure: 18%
- High releases (partly early): 9%

(SSK 2004)

The difference regarding unfiltered releases (15% for EPR, 27% for the KONVOI plant) is less than a factor of 2.

It appears to be the main advantage of the measures for core melt mitigation implemented at the EPR to avoid filtered venting, which would otherwise occur in about half of the CD cases and leads to rather small releases, apart from noble gases.

Regarding probabilistic design targets and PSA results, it is remarkable that they do not fully cover all accident sequences in the most recent and most comprehensive documents published on the EPR by the reactor designer (UK-EPR 2008).

For core damage frequency, there are design targets for the whole spectrum of initiators (internal events, internal hazards, external hazards), and corresponding results.

For the Plant Damage States involving loss of containment and for large early releases, however, there is a probabilistic design target for internal events only. Internal and external hazards are not covered for those cases; accordingly, results of the PSA level 1+ for Plant Damage States are provided solely for internal events.

This reduces the significance of the PSA results. Internal and external hazards contribute about 55% to overall CDF and it can be expected that they also contribute substantially to the frequencies of Plant Damage State with containment loss, and of sequences leading to large, early releases.

The PSA results reported here differ from results published earlier for the Olkiluoto-3 EPR. According to the regulatory authority's Safety Assessment for Olkiluoto-3, the CDF from internal events is about 1.44E-6/yr (STUK 2005), more than twice as high as the results for the UK-EPR. The CDF for internal events does not depend significantly on the site selected; hence it appears difficult to explain this difference.

Ex-vessel cooling of molten core

Concepts of ex-vessel cooling are to be applied in most of the reactor types listed in the EIA Report – including the ESBWR. For basic problems associated with the core catcher see section 5.2.1.1 Core Catcher of the present statement.

Containment vulnerability to aircraft crash

For the UK-EPR as well as the EPR being built at Flamanville in France, the risk of aircraft crash has been assessed on the basis of French regulatory requirements (UK-EPR 2008, 2.R.4). Those regulations cover accidental crashes and do not require protection against the crash of a commercial airliner (WISE 2001).

In Finland, new nuclear power plants are to be designed against the crash of a large passenger aircraft. Such an event must not cause damage which would lead to an immediate significant radioactive release, and it must be possible that the most important safety functions can be started and maintained in this case with adequate certainty (STUK 2005).

For the Olkiluoto-3 EPR, it was reported in 2005 that analyses taking into account varying loads up to the Airbus A380 suggested that at least no major modifications to the building dimensions were required. However, the analyses were incomplete at that time.

Apart from the direct effects on the building structure, an impact will also cause vibrations inside buildings which might threaten the integrity of systems and equipment. Vibration calculations were performed and have to be revised during construction.

All in all, regarding the impact of a commercial airliner, the authority reported that numerous details still need to be finalized, analyses need to be completed, and the results of the analyses must be verified experimentally (STUK 2005).

To the knowledge of the authors of this statement, there are no more recent publications regarding passenger aircraft crash at Olkiluoto-3.

The case of the 2A-LOCA

The double-ended guillotine break (2A-LOCA) is not a design basis accident for the UK-EPR and for the Flamanville EPR, because of the break preclusion assumption. It is dealt with in the context of risk reduction (RRC-C, see above).

Therefore, the analysis of the 2A-LOCA is carried out using best-estimate criteria and assumptions, and not on a conservative basis as appropriate for DBAs. For example, the single failure criterion is not applied and no system unavailability due to maintenance is assumed (UK-EPR 2008, 2.S.3.2a). Hence, the redundancy of the emergency core cooling system in this case is only n+1, and not n+2 as required for DBAs.

However, for the Olkiluoto 3 EPR, it is reported that analyses have shown that there is in fact n+2 redundancy of the ECCS for the 2A-LOCA, although the ECCS and the residual heat removal system conform to the original design, i.e. were not upgraded (STUK 2005). This statement appears surprising since it implies that in fact, no special treatment in the context of risk reduction would be required for the 2A-LOCA at the EPR and the 2A-LOCA could be included in the DBAs.

4.2.3 Extended Discussion about ESBWR

4.2.3.1 Special features of ESBWR's

The four key attributes of the ESBWR, compared to Generation II BWRs, are the following – according to the designer (HINDS 2006):

- Simplification (reduced systems and structures, passive safety systems);
- Standardized design (seismic design envelope for all site conditions, standardized components);
- Operational flexibility (increased operating margins);
- Improved economics (reduced materials and buildings, reduced and simpler systems, reduced construction time, features from ABWR used).

The designer emphasizes, on the one hand, that the ESBWR is equipped with numerous design features that contribute to a low core damage frequency; on the other hand, it is pointed out that the improvements are based on the operating experience of earlier BWR types, most notably the ABWR.

The design modifications which are to reduce risk include, as presented by the designer (UK-ESBWR 2008):

- Front-line safety systems are passive and therefore have less reliance on the performance of supporting systems or operator actions. ESBWR does not require operator action for successful mitigation until 72 hours after the onset of an accident.
- The design reduced reliance on AC power by using 72 hour batteries for several components. The core can be kept covered without any AC sources for the first 72 hours.
- There is no recirculation system; thus, frequency and consequences of loss of coolant accidents are reduced.
- The design reduces the possibility of a LOCA outside containment by designing practically all piping systems, major components and sub-systems connected to the reactor coolant pressure boundary (RCPB) to a rupture strength at least equal to full RCPB pressure.
- A highly reliable passive containment cooling system (PCCS) significantly reduces the probability of loss of containment heat removal.
- The ESBWR containment has a higher safety factor than earlier BWRs; it is to minimise the effects of direct containment heating, ex-vessel steam explosions and core-concrete interaction.

Ex-vessel cooling of molten core (core catcher) and other measures for core melt mitigation

In the following, a brief description of the ex-vessel cooling concept based on documents of the reactor designer is provided (UK-ESBWR 2008, 19.3.2).

The lower drywell floor of the ESBWR is designed with sufficient space to enhance core debris spreading and contains the BiMAC (Basemat-internal Melt Arrest and Coolability) device which together with the GDCS (Gravity Driven Cooling System) is to protect containment liner and basemat, and to guarantee long-term coolability and stabilization of core debris.

The BiMAC device consists of pipes forming a jacket, covered with a sacrificial refractory layer and a steel plate to protect it from falling control rod drive housings (this plate will rapidly be penetrated by the melt). The entire volume corresponds to 400% of the debris from the full core. (Not all dimensions of the core catcher are provided in the source; those that are, are subject to change since they are for conceptual design purposes only).

When the characteristic high temperature profile of a core debris discharge is detected, the lower drywell deluge subsystem of the GDCS is automatically activated. Thus, it is to be practically guaranteed that flooding only occurs after the discharge of core material. Flooding prior to discharge could lead to steam explosions; the hazard of this is seen as remote. The deluge system fills the pipes of the BiMAC jacket and subsequently floods the spread core debris.

No details concerning analyses and investigations of the functioning of the core catcher are provided in the source.

The system appears to be roughly comparable to the EPR core catcher, with one important difference: The ESBWR core catcher does not include the feature of temporary melt retention in the reactor pit; the core debris directly falls into the BiMAC device as soon as the RPV is molten through.

Further measures for core melt mitigation to be implemented at the ESBWR are (UK-ESBWR 2008, 19.3.2):

- Prevention of hydrogen detonation, mostly relying on inertisation of the containment. The time required for the oxygen concentration to reach 5% in case of a core melt accident is reported to be significantly greater than 24 hours.
- Prevention of high pressure melt ejection, employing a highly reliable depressurization system.
- The possibility of filtered venting of the containment is implemented; however, it is reported that venting will not be necessary for at least 24 hours in an accident scenario in which containment heat removal is lost.

These features, however, are not particular to Generation III plants. They are also implemented in Generation II BWRs, even in older plants.

4.2.3.2 Probabilistic design targets and PSA results of ESBWR's

This section summarizes the most recent information published on probabilistic safety objectives and PSA results for the ESBWR (UK-ESBWR 2008).

Probabilistic safety goals

The PSA results are compared against the US N.R.C. probabilistic goals (which do not constitute regulatory requirements) (UK-ESBWR, 19.1.2):

- Core damage frequency (CDF) less than 1E-4/r.yr
- Large release frequency (LRF) less than 1E-6/r.yr
- Conditional containment failure probability (CCFP) less than approximately 0.1 of the composite of all core damage sequences assessed in the PSA.

PSA results

Table 12: Preliminary core damage frequency estimates (UK-ESBWR 2008, 2.6.6).

Category	CDF/yr
at-power internal events	1.22E-8
at-power fire	1.21E-8
at-power flood	3.7E-9
shutdown internal events	8.8E-9
shutdown fire	2.32E-8
shutdown flood	1.6E-9
Total	**6.16E-8**

Main contributors for internal events at-power are:

- Transient with inadvertent opening of a safety relief valve: 36.5%
- Loss of feedwater and condensate: 18.69%
- General transient: 18.4%
- Loss of preferred (off-site) power: 11.54%
- LOCA: 8.66%
- Loss of decay heat removal: 3.75%
- Line break outside of containment: 2.5%.

The external events models are presently under revision; hence, there are no results for external hazards provided in the source.

No explicit values for large release frequencies are provided in the available sources. However, the probability of receiving a dose greater than 0.25 Sv at 805 m from the reactor is reported to be 2E-9/r.yr (internal events only). This could be taken as an indicator that, according to the PSA results, LRF is below 1E-8/yr.

4.2.3.3 Indications for potential weak points and problems of ESBWR‘s

General discussion

The ESBWR is presented as an innovative reactor type of Generation III; at the same time, it is emphasized that processes and technologies from the already developed and operationally proven ABWR are being utilized (HINDS 2006).

In the context of safety, the new features of the ESBWR are particularly accentuated. On the other hand, the reliance on the proven ABWR design is underscored in the economic context – as making construction schedules and cost planning more dependable.

It is noteworthy that the ESBWR is to rely on natural circulation and passive safety system much more than the ABWR; while having a power output which is increased by nearly 15 percent, and a reactor pressure vessel with the same diameter.

In principle, passive safety is easiest to realize with low power density. Higher power density will reduce safety margins. Even if passive safety systems are generally preferred for controlling emergency situations, they have a disadvantage, which should not be neglected: In any case, there is a basic problem associated with passive safety systems: There is the danger of insufficient possibilities for intervention in case of unforeseen problems. This danger will be more marked with smaller margins.

Ex-vessel cooling of molten core

Concepts of ex-vessel cooling are to be applied in most of the reactor types listed in the EIA Report – including the ESBWR.

The basic problems associated with the core catcher are treated in section 5.2.1.1 Core Catcher.

There is one marked difference between the ESBWR and the EPR core catcher concept. The EPR concept includes temporary melt retention in the reactor pit, before the molten core reaches the spreading area (the core catcher proper). The objective of the melt retention is to make the subsequent spreading and stabilization of the corium largely independent of the uncertainties of in-vessel pool formation and RPV failure.

This issue is not discussed in the available references for the ESBWR.

The ESBWR core catcher simply consists of a device located at the lower drywell floor (BiMAC device), directly underneath the reactor pressure vessel. This could lead to specific problems; for example, if the BiMAC device is flooded after discharge of a part of the core debris, and more debris follows later and falls into the flooded area, giving rise to the hazard of steam explosion.

Vulnerability to external hazards

Little information relevant for the vulnerability to external hazard is contained in the references. In particular, there are no data on containment wall thickness. It is only stated that the containment is a reinforced concrete cylindrical structure with an internal steel liner (UK-ESBWR 2008, 6.2.1).

In the context of external hazards, it is emphasized that protective features *"can be described on a need to know basis under appropriate safeguard controls and approvals"* (UK-ESBWR 2008, 2.7.2). While it is appropriate not to publish details of protective systems which could be useful for the planning of an attack, it is unclear why basic containment features like wall thicknesses, which are being published for other NPPs (including the EPR) could not also be provided for the ESBWR.

It is mentioned that the condensers of the passive containment cooling system (PCCS) are located outside the containment (UK-ESBWR 2008, 6.2.2). This constitutes a potential weak point in case of external attacks.

4.3 Conclusion

The EIA Report does not provide sufficient information about the reactors considered for LO 3. From the presentation in the EIA, an evaluation of safety, the maximal source term and the probability of occurrence is not possible.

There is very little operational experience with the reactor types listed in the EIA. Only one of those types (ABWR) has been in operation so far – and only as an earlier variant of the type marketed today.

The majority of the reactor types listed relies on ex-vessel cooling (i.e., installation of a core catcher) for the control and mitigation of severe accidents. Fundamental problems still remain regarding the reliable functioning of a core catcher, e.g. concerning interactions between molten core and concrete, high uncertainties regarding heat transfer rates, cracking of the concrete surface, complications due to H_2 generation, insufficient cooling with water flooding only, or steam explosions. There are no recent in-depth discussions on these topics. This could imply that no substantial progress has been achieved in solving the problems.

The concept of in-vessel cooling is basically more promising, but difficult, if not impossible, to successfully implement in large reactors.

PSA results for the reactor types can be found in the open literature. Core damage frequencies span almost two orders of magnitude. The reactor types at the upper limit (EPR, APR-1400) lie in the same range as reported for newer LWRs of Generation II. It remains unclear, whether the Finnish probabilistic target of 5E-7/yr for large releases covers all accident initiators – internal events, internal hazards and external events.

Regarding the impact of a commercial airliner on an EPR, the authority reported that numerous details still need to be finalized, analyses need to be completed, and the results of the analyses must be verified experimentally (STUK 2005).

The two reactor types selected for extended discussion – the EPR and the ESBWR – are representative for the spectrum of reactors under consideration for LO 3. First of all, one PWR and one BWR were chosen. Furthermore, and more important, the two reactor types differ considerably regarding the degree of technological innovation they incorporate.

The EPR has been developed from the German KONVOI and the French N4 PWR reactors. Like the precursor types, it depends mostly on active safety features. Compared to those earlier types, there is only one significant new feature: The core catcher, for ex-vessel corium cooling. The EPR's core melt frequency lies in the same range as the CDF reported, for example, for newer German PWRs. This implies that another new feature of the EPR, the in-containment refuelling water storage tank, does not have a significant effect on overall plant safety.

According to PSA results available, the reduction of the large release frequency due to the core catcher, again compared to newer German PWRs, is less than a factor of two. The main advantage of the measures for core melt mitigation at the EPR seems to be that filtered venting can be avoided, which would otherwise occur in about half of the CDF cases.

The ESBWR, on the other hand, is among the most advanced new designs listed in the EIA report. It is developed from another Generation III BWR (the ABWR) and is equipped with numerous new design features. It relies, to a considerable extent, on passive safety systems – all front-line safety systems are passive, as well as the containment cooling system needed in case of a severe accident. ESBWR does not require operator action for successful mitigation until 72 hours after the onset of an accident.

The ESBWR is also equipped with a core catcher, with a simpler design than that of the EPR.

The core damage frequency of the ESBWR, as reported, is the lowest of all the reactor types listed, considerably lower than CDFs of Generation II reactors. The large release frequency appears to be well below 1E-8/yr, according to a PSA level 1+, and hence is also considerably lower than that of Generation II reactors.

The flipside of the ESBWR's apparently high safety is this reactor type is still in an early planning stage. No ESBWR units are in operation, under construction or firmly planned yet. Hence, PSA results might be less reliable than results for further developed types.

To the knowledge of the authors of this statement, no results of level 2 PSAs, particularly no values for large release frequencies, have been published for the ESBWR. Also unknown is the assessment of the Finnish side of the general vulnerability of the ESBWR to external hazards, particularly to attacks. It should be clarified, whether the Finnish side is able to provide the information.

It is noteworthy that the ESBWR with its natural circulation and passive safety system on one hand, but relatively high power density on the other hand will provide reduced safety margins. The danger of insufficient possibilities for intervention in case of unforeseen problems due to passive safety systems will be more marked the smaller the margins are.

In the view of the authors, it is necessary to assess all the reactor types listed in the EIA Report according to their place in the scale reaching from “basically tested design, not many new features” to “new, largely untried design with many new features”. (In most cases, new features will imply the use of more passive safety systems.) It is likely that the EPR will be found at one end of this scale, and the ESBWR at or near the other.

An assessment of this kind would permit a meaningful comparison between reactor types. Without such an assessment, comparisons between safety features and PSA results of different reactor types are of limited significance. It does not appear as admissible to simply regard reactors of different types as “black boxes”, in spite of fundamental differences regarding the extent of new features incorporated in the design.

5 SAFETY AND ACCIDENT ANALYSIS

5.1 Treatment in the EIA Report

The EIA Report states, that the EIA procedure of the new power plant unit will be followed by a Government resolution application in accordance with the Nuclear Energy Act, if the project will be continued. "When applying for the resolution, the plant supplier has not yet been chosen, so the content of the safety report will focus on the safety assessment described in the Decision by Government regarding nuclear safety (State 395/91) and in the YVL Guides of the Radiation and Nuclear Safety Authority regarding design." And further: "When applying for a construction licence in accordance with the Nuclear Energy Act, the plant supplier has been chosen, and detailed plant type-specific safety assessments will be supplied to the authority" (FORTUM 2008, 151)

Chapter 12 of the EIA Report (FORTUM 2008) in a general manner deals with the safety of the new NPP at Loviisa. General principles like "defence in depth" and "multiple barriers" as essential nuclear safety principles are mentioned.

"The new power plant unit is designed such that it fulfils the requirements set by the authorities. High safety goals are set for the power plant unit and the design also takes into consideration severe accidents caused by core melting. It is required of the new power plant unit that possibility of an accident leading to core melting occurs is less often than once in 100,000 years and large environmental radioactive releases are possible less often than once in 2,000,000 years. In addition to being prepared for a severe accident, the new plant unit must also be protected against external threats and terrorism" (FORTUM 2008, 150).

Impacts of anticipated operational transients and postulated accidents are treated in chapter 12 of the EIA Report (FORTUM 2008, 146). The dose limits for the population are given as follows:

- For an anticipated operational transient (with a frequency of occurrence of 1E-2 and even higher): 0,1 mSv/yr
- For postulated accidents Class 1 with a frequency of occurrence of < 1E-2: 1 mSv/yr
- For postulated accidents Class 2 with a frequency of occurrence of < 1E-3: 5 mSv/yr
- For events handled as extension of postulated accidents, i.e. a common cause failure or a complex combination of failures without severe fuel damage: 20 mSv/yr.

Requirements for Severe Accidents

The Decision of the Council State (State 395/91) requires that "a severe reactor accident shall not cause early adverse health effects to the residents in the plant vicinity or long-term restrictions on the use of extensive land or water areas" (FORTUM 2008, 153). Fulfilling the requirements of (State 395/91) includes proving that the frequency of exceeding this limit is extremely small, i.e. below 5E-7 per year.

For exemplary dose calculations, the source term for the severe accident is assumed to be 100 TBq Cs-137 (plus a corresponding proportion of other caesium isotopes), 1,500 TBq I-131 (plus a corresponding proportion of other iodine isotopes) and 100% of noble gases, e.g. Krypton and Xenon. Release height is assumed to be 100 m through the ventilation stack (FORTUM 2008, 153).

Calculation of Radiation Doses

With this source term the radiation dose for the population has been assessed for the EIA Report. The radiation dose caused by a release has been divided into two parts, one dose during the first 24 hours of radiation exposure, a second dose during the subsequent 50 years. Following computer programs have been used for deposition and radiation doses assessment: LENA95 (BÄVERSTAM 1996), TRADOS (NORDLUND et al., 1985) and TUULET (SAIKKONEN 1992). TUULET was used for assessing the deposition and radiation doses caused to an adult at short distances (100 km) – both during 24 hours and the subsequent 50 years. Results of the program LENA95 (BÄVERSTAM 1996), fitted to the results of the TUULET program, were used for assessing the deposition and radiation doses during the first 24 hours at medium distances (300 km). Radiation doses at distances of 300 km, 500 km and 1,000 km during the subsequent 50 years, as well as depositions and radiation doses during the first 24 hours at distances of 500 and 1,000 km, have been "estimated on the basis of the distance dependence of the heaviest radiation doses assessed with the TRADOS program" (FORTUM 2008, 154).

The location of the measuring point for gathering weather data (used for the dispersion calculation) is given. The EIA Report states, that for calculation of the radiation doses of the first 24 hours those weather conditions were used, in which the radiation dose caused by the release plume is "as massive as possible" (FORTUM 2008, 154).

Sea level changes

On October 16th 2007, the Ministry of Trade and Industry gave its statement on the EIA programme of the project. Within the statement, a summary of comments, opinions and statements from various ministries, experts, authorities and others is given.

Among those, the Ministry of Agriculture and Forestry suggests an "assessment of threats posed by changes and rise in the sea water levels because these have an impact on the reliability and safety of the nuclear power plant" (FORTUM 2008, Appendix1,197). The Statement of the coordinating authority on the EIA programme and its consideration (FORTUM 2008, 32) presents matters which, according to the statement of the Ministry must be taken into account when compiling the assessment report and in the meanwhile "have been gone through point by point". Concerning future sea level changes it is stated: "Sea level changes ... have been described in Chapter 7. Sea level changes, impacts of climate change and other external threats are taken into consideration in the design of the power plant unit" (FORTUM 2008, 33/34).

Notification in case of emergency

Within the statement on the EIA programme of the project given by the Ministry of Trade and Industry, the "Innenministerium Mecklenburg-Vorpommern (Germany)" proposes taking into consideration "a description of how Germany, among other countries, will be informed in case of emergency" (FORTUM 2008, 208).

5.2 Discussion

5.2.1 Requirements for Severe Accidents

5.2.1.1 Frequency

The limits for severe accidents are laid down in the Decision of the Council State, 395/91. A new Government Decree replacing 395/91 is at present in the draft stage. However, the limits for severe accidents will not be changed.

Besides the Finnish YVL guides the European Utility Requirements are mentioned.

The 395/91 specifies the following (STUK 2005):

"The limit for the release of radioactive materials arising from a severe accident is a release that causes neither acute harmful health effects to the population in the vicinity of the nuclear power plant, nor any long-term restrictions on the use of extensive areas of land and water. For satisfying the requirement applied to long-term effects, the limit for an atmospheric release of caesium-137 is 100 TBq. The combined fall-out consisting of nuclides other than caesium-isotopes shall not cause, in the long term, starting three months from the accident, a hazard greater than would arise from a caesium release corresponding to the above-mentioned limit. The possibility that, as a result of a severe accident, the above-mentioned requirement is not met, shall be extremely small."

The EIA Report states, that "the Loviisa-3 nuclear power plant will be designed in compliance with the instructions issued by the Radiation and Nuclear Safety Authority (YVL Guides)" (FORTUM 2008, Table 5-1). According to the Radiation and Nuclear Safety Authority's Guide YVL 2.8, the expectation value for the frequency of a Caesium-137 release exceeding 100 TBq assessed by a PSA should be below 5E-7/yr (STUK 2005).

5.2.1.2 Source Term

The source term assumed for exemplary dose calculations in the EIA Report (FORTUM 2008, 153) is considerably smaller than the term for a worst-case accident with early containment failure (which would be in the order of magnitude of 300,000 TBq Cs-137 and 1,800,000 TBq I-131). Very effective measures of accident mitigation are required to reliably achieve such a comparatively low source term.

The MTI demanded in its statement on the scoping procedure[8] of the EIA to present the safety planning criteria of the prospective plant with respect to the limitation of radioactive emissions as well as an assessment of how the safety requirements in force will be met. Fortum has ignored this very important requirement of the MTI.

In relation to the Cs-137 releases, the estimated releases of I-131 (FORTUM 2008, 153) appear to be rather small (at least for a PWR; the situation for a BWR could be different).

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[8] "… the safety planning criteria for the prospective plant must be presented with respect to the limitation of emissions of radioactive substances and environmental impacts, as well as an assessment of the possibilities of meeting the safety requirements in force." (MTI 2007, 210)

For example, for German PWRs, considerably higher releases of I-131 are reported for accident scenarios with comparable Cs-137 releases (SSK 2004):

- DRS Phase A, release category "AF-Leckage ND*": Released amount of I-131 higher by a factor of 125 than the release of Cs-137;
- PSA level 2 for GKN-2 – release category FKE: Release of I-131 higher by a factor of 55–1,400 than the release of Cs-137.

It is likely that special mitigation measures would be required to reduce iodine releases. There is no discussion of such measures in the EIA Report.

Furthermore, it is questionable whether the restriction to Cs-137 (and other Cs isotopes), I-131 (and other I isotopes) and noble gases is sufficient, even for a rough assessment of accident consequences. In the "Criteria for Limited Impact" of the European Utility Requirements, nine nuclides are listed to be taken into account when determining whether the criteria are fulfilled (EUR 2001). For further information also see the annex to this chapter.

Taking the source term from the EIA Report (while keeping in mind that the consistency and sufficiency of this source term are questionable), it is possible to check whether the EUR criteria are kept.

There are four Criteria for Limited Impact:

1. No emergency protection action beyond 800 m
2. No delayed action beyond 3 km
3. No long-term action beyond 800 m
4. Limited economic impact.

For the criteria 1–3, the releases of nine nuclides have to be multiplied with weight coefficients and added. The sums have to be below a given limit. There are different coefficients for elevated release (from stack, release height 100 m or more) and for ground releases (release height below 100 m).

For criterion 4, the maximal permissible release for each of three reference isotopes (I-131, Cs-137 and Sr-90) is specified.

The sums for criteria 1–3 can be calculated with the source term of the EIA Report and divided by the limits according to EUR.

For criterion 1, the limit is exceeded by the source term – by a factor of 10 for ground release, and a factor of 1.8 for elevated release (for a combination of the two release forms, the factor will be between those two values). For criteria 2 and 3, the weighted sum is well below the limit – at least by a factor of 15.

For criterion 4, the limit for Cs-137 is exceeded by a factor of 3.3.

(For more details of the comparison of the EIA Report source term with EUR criteria, see the annex to this chapter.)

5.2.1.3 Frequency of exceeding the limit

According to EUR, the expected frequency of a release higher than specified in the Criteria for Limited Impact has to be 1E-6/yr.

The frequency limit according to Finnish regulations is half of this value. Taking into account the uncertainties and limitations of PSAs (HIRSCH 2006), this difference appears to be of little significance.

5.2.2 Calculation of Radiation Doses

The weather conditions used for calculation of radiation doses are not explained sufficiently. Furthermore, there is no information on what type of dispersion model is used by the computer programs LENA95, TRADOS and TUULET. The assessment of doses up to a distance of 1.000 km is based on estimations, which have not been explained within the EIA Report. The radiation dose values in table 13.1 (FORTUM 2008, 154) are presented without systematic errors. This could imply that no error calculation has been carried out. In general, a value is meaningless without its error. Furthermore, it is to be assumed that, due to uncertainties, every extrapolation and estimation effects in an increase of the systematic error of the value. The calculation of errors is not provided within the EIA report at all. Therefore the results in table 13.1 (FORTUM 2008, 154) are irreproducible and not comprehensible. However it is unclear whether one of these programs is appropriate for modelling long-range transport and deposition of radionuclides.

5.2.3 Sea level changes

The Ministry of Agriculture and Forestry suggests an assessment of threats posed by changes and rise in the sea water levels (FORTUM 2008, Appendix 1, 197).

The expected extent of sea water level changes has been assessed in chapter 7 of the EIA Report: up to 60 cm till the year 2100. Unfortunately the suggested assessment of possible threats posed by the expected rise in the sea water level has not been performed.

5.3 Conclusion

Concerning safety and accident analysis, Austria’s main interests are to assess a possible future impact on Austria’s territory caused by accidental radioactive releases from the LO 3 NPP, and to develop a catalogue of countermeasures.

The MTI demanded in its statement on the scoping procedure of the EIA to present the safety planning criteria of the prospective plant with respect to the limitation of radioactive emissions as well as an assessment of how the safety requirements in force will be met. Fortum has ignored this very important requirement of the MTI. It is recommended that the responsible authorities claim the planning criteria and evaluation of reaching the targets.

In the EIA Report a dose assessment of the accidental release with the exemplary source term is presented for a region of 100 km from the NPP. For a distance up to 1,000 km the dose assessment is based on extrapolation and estimations. All dose values are published without systematic errors.

Regarding releases of Cs-137, this source term is based on Finnish regulations (100 TBq). This corresponds to an accident sequence which is controlled and mitigated to a considerable degree; in case of a severe accident with early containment failure, the amount of Cs-137 released could be higher by a factor of 1,000 and more.

Although there is some arbitrariness involved in selecting the Cs-137 source term for a mitigated accident, the order of magnitude chosen appears to be appropriate. At least an explanation for choosing this source term would be of interest. It is important, however, to keep in mind that accidents with much more severe releases cannot be excluded for all the reactor types listed in the EIA Report.

For an assessment of the potential impact of a severe accident at the LO 3 NPP the information of the EIA Report is not sufficient. A worst case has to be discussed and the source term has to be proved to be the maximum release. Sufficient information about the used weather conditions and the type of dispersion model used by the computer programs have to be provided. The results in table 13.1 (FORTUM 2008, 154) shall be published together with their calculated errors.

For modelling the long-range transport, diffusion and deposition of radionuclides, more sophisticated tools are required, such as the Lagrangian particle dispersion model FLEXPART. FLEXPART is a model suitable for the meso-scale to global-scale calculations, which is freely available and used by many groups all over the world (STOHL 1998). However it is unclear whether one of the programs used for dose assessment according to the EIA report is appropriate for modelling long-range transport and deposition of radionuclides.

The source term as assumed in the EIA Report for exemplary dose calculations appears questionable regarding the relation between Cs-137 and I-131; and it does not take all nuclides which are required for checking EUR release criteria (Criteria for Limited Impact) into account. Even so, not all EUR criteria are kept by the source term in the EIA Report. Criterion 1 (no emergency protection action beyond 800 m) is not fulfilled; Criterion 4 (limited economic impact) is not fulfilled for the release of Cs-137.

Within the scope of this expertise, it cannot be discussed whether it can be expected that the reactor types listed in the EIA Report conform to Finnish regulations. A detailed technical assessment of the reactors would be required to answer this question. At least the Finnish authorities should provide a comprehensive list of the specific safety requirements for Generation III reactors in Finland.

It cannot be determined in how far threats posed by the assessed rise in the sea water levels do exist. An assessment of threats posed by changes and rise in the sea water levels, as suggested by the Ministry of Agriculture and Forestry, would be required to answer this question.

Last but not least we suggest taking into consideration a description of how Austria, among other countries, will be informed in case of emergency.

ANNEX Explanation of Criteria of Limited Impact

The following table shows the nine nuclides which are taken into account by the EUR when determining whether the Criteria for Limited Impact are kept, as well as the weight coefficients for ground and elevated releases, for Criterion 1.

Below is the formula being used to determine acceptance.

Isotope group	Coefficients for ground level releases C_{ig}	Coefficients for elevated releases C_{ig}
Xe_{133}	$6{,}5.10^{-8}$	$1{,}1.10^{-8}$
I_{131}	$5{,}0.10^{-5}$	$3{,}1.10^{-6}$
Cs_{137}	$1{,}2.10^{-4}$	$5{,}4.10^{-6}$
Te_{131m}	$1{,}6.10^{-4}$	$7{,}6.10^{-6}$
Sr_{90}	$2{,}7.10^{-4}$	$1{,}2.10^{-5}$
Ru_{103}	$1{,}8.10^{-4}$	$8{,}1.10^{-6}$
La_{140}	$8{,}1.10^{-4}$	$3{,}7.10^{-5}$
Ce_{141}	$1{,}2.10^{-3}$	$5{,}6.10^{-5}$
Ba_{140}	$6{,}2.10^{-6}$	$3{,}1.10^{-7}$

The acceptance criterion is that:

$$\sum_{i-1}^{9} R_{ig} * C_{ig} + \sum_{i-1}^{9} R_{ie} * C_{ie} < 5 * 10^{-2}$$

Criteria 2 and 3 are following the same scheme, whereas Criterion 4 specifies the limits for three reference nuclides, each of which must not be exceeded (4,000 TBq for I-131, 30 TBq for Cs-137 and 400 TBq for Sr-90).

6 NUCLEAR FUEL PRODUCTION

6.1 Availability of Uranium

6.1.1 Treatment in the EIA Report

Regarding availability of uranium the EIA Report states, that currently the global annual demand for uranium concentrate (U_3O_8) is some 65.000 tonnes. At the moment, the production of new natural uranium covers about 60%–70% of the demand. The rest is covered by emptying storages, by reprocessing spent fuel and by using depleted weapons-grade uranium.

"Owing to the common occurrence, uranium resources will be enough far into the future. The adequacy of uranium resources depends on the cost level of economically profitable uranium production. The more expensive the alternative sources of energy are, the more money can be used for uranium fuel production and the larger the available uranium resources are. ... The amount of uranium needed for nuclear power production has been estimated to grow to 81.000 tonnes by 2020 and to 110.000 tonnes by 2030. At these consumption levels, the uranium resources will be enough for several hundred years" (FORTUM 2008, 162).

"The fuel assemblies of the existing power plant units in Loviisa are both Russian (TVEL) and British (Westinghouse/BNFL). The uranium used in the assemblies of both manufacturers originates in Russia" (FORTUM 2008, 166). With regard to the production volume, the largest operating uranium mine in Russia is located in the Priargunsk (former Krasnokamensk) mine area (FORTUM 2008, 164).

6.1.2 Discussion

Contrary to the hypothesis, that the uranium resources will be enough for several hundred years, the analysis of data on uranium resources, as discussed in the paper EWG series No 1/2006 of the Energy Watch Group (EWG 2006), leads to the assessment that discovered reserves are not sufficient to guarantee the uranium supply for more than thirty years.

Production of new natural uranium

„Eleven countries have already exhausted their uranium reserves. In total, about 2.3 Mt of uranium have already been produced. At present, only one country (Canada) was left having uranium deposits containing uranium with an ore grade of more than 1%, most of the remaining reserves in other countries have ore grades below 0.1% and two thirds of reserves have ore grades below 0.06%. This is important as the energy requirement for uranium mining is at best indirect proportional to the ore concentration" (EWG 2006).

In Priargunsk/Krasnokamensk (Russia), where Fortum procures uranium from, the extraction of uranium today is mainly carried out in underground mines, where the uranium content of ore is 0.3–0.4% (FORTUM 2008, 164). In Olympic Dam, the Australian mine where the Finnish company Teollisuuden Voima Oy (TVO) procures uranium from, the ore grades are even as low as 0.044% (BOSSEL 2007) to 0.053%

(DIEHL 2006). To extract 1 t of uranium out of 1% ore containing material requires the processing of 100 t. Extracting the same amount from 0.01% ore requires the processing of 10.000 t.

At the annual demand of 2006, the proved reserves (below 40 $/kg U extraction cost) and stocks are going to be exhausted within 30 years. Possible resources (which contain all estimated discovered resources with maximal 130 $/kg extraction costs) will be exhausted within 70 years. There are problems and delays with the biggest new mining projects (e.g. disastrous water brake-in at the mines of Cigar Lake in Canada), which are causing doubts whether these extensions can be realized at all. If not, then even before 2020 supply problems are likely. Otherwise, if all estimated known resources up to 130 $/kg U extraction cost can be converted into production volumes, a shortage can at best be delayed until about 2050 (EWG 2006).

Emptying stockpiles

Only 42.000 t/yr of the 67.000 t/yr current uranium demand are supplied by new production, the rest of some 25.000 t/yr is drawn from stockpiles. The stock consists of stocks at mines, reactor sites, conversion of nuclear weapons and reprocessing of nuclear waste. While in 2002 the amount of uranium in stocks was estimated to be some 390.000–450.000 tonnes, it should be reduced to about 210.000 tonnes of uranium or even less by the end of 2005. These stockpiles have been accumulated before 1980 and will be exhausted within the next 10 years. Therefore uranium production capacity must increase by at least some 50% in order to match future demand of current capacity (EWG 2006).

Reprocessing of spent fuel

The reprocessing of nuclear fuel comes together with nuclear weapons proliferation risk. „The technology of reprocessing is described in open literature, and there was sufficient open literature on nuclear weapons even in the mid-1960ies to allow three graduate students in the US to successfully design an implosion weapon with a 15 kiloton yield with two man-years of effort. No other bulk electrical energy or process heat source (coal, oil, natural gas, hydroelectric power, wind power, solar power, biomass, etc.) has such proliferation concerns associated with it." (BMLFUW 2007). With a focus on the commercial nuclear fuel cycle, reprocessing of spent fuel is one of the four principal points of vulnerability for nuclear weapons proliferation.

In the following figure the relation of possible uranium production profiles, reported reserves and resources and the annual uranium fuel demand of reactors are shown, where the "reference scenario" represents the most likely development, the "alternative policy scenario" represents the scenario based on policies to increase the share of nuclear energy with the aim of reducing carbon dioxide emissions. The calculations are based on data from the Nuclear Energy Agency, the forecasts are based on the 2006 scenarios by the International Energy Agency (EWG 2006).

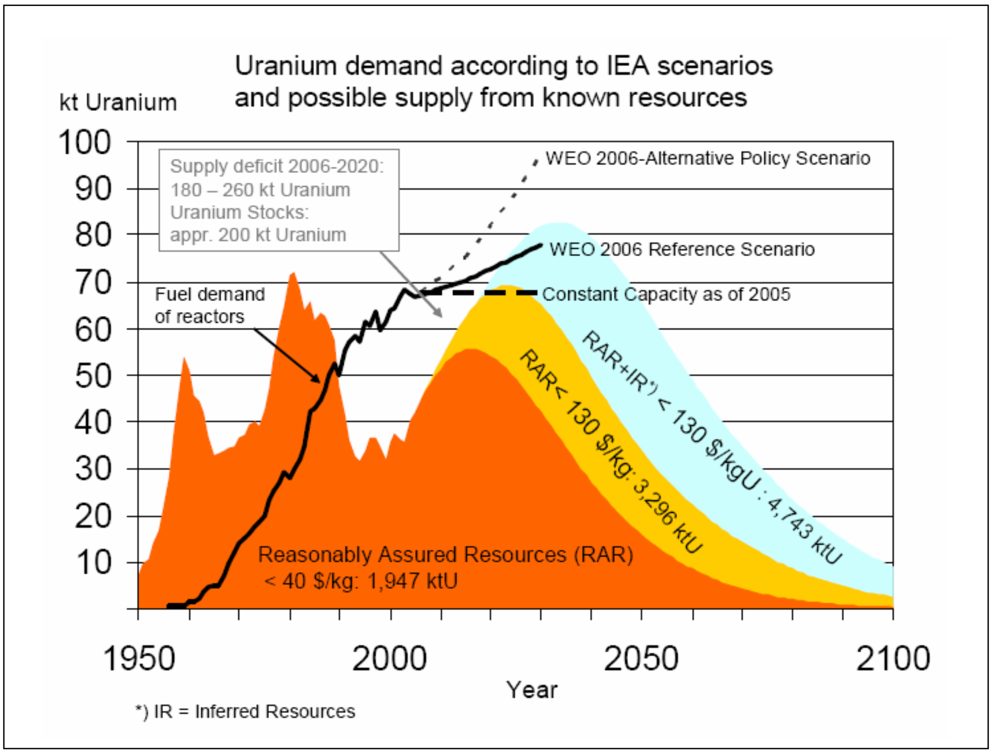

Figure 2: Past and projected uranium production. The black line shows the fuel demand of reactors currently operating together with the latest scenarios in the World Energy Outlook 2006 of the IEA.

6.1.3 Conclusion about availability of Uranium

While summarizing the facts, we have to assess that the availability of uranium is awfully well an obstacle for continuing or expanding the use of nuclear power.

New uranium production will require a much higher price level. Vadim Zhivov, Chief Executive Officer of Russia's state-owned mining company Uranium Holding ARMZ, stated on April 28th, 2008: „Priargunsk, Russia's only listed uranium mining company, will raise prices by 15 percent this year" (HUMBER 2008).

As stated in the assessment of the Austrian nuclear advisory board (BMLFUW 2007) – new uranium production is also a matter of proliferation risk and is definitely not a CO_2-free technology.

6.2 IMPACTS OF NUCLEAR FUEL PRODUCTION

6.2.1 Treatment in the EIA Report

Regarding impacts of nuclear fuel production the EIA Report states, that Fortum's nuclear fuel procurement takes environmental impacts into consideration as early as when calling for bids. "The bidders are expected to present their environmental management systems in the bids or to describe how the environmental impacts caused by their operations have been taken into consideration. When comparing the bids, it is estimated whether the operation is proper and adequate in respect of the legislation of the supplier's country. Fortum carries out regular audits of the quality management systems of its fuel suppliers. ... Fortum also controls regularly the manufacture of fuel assemblies in the fuel factories" (FORTUM 2008, 166).

The EIA Report states that in Russia Fortum procures uranium from the Priargunsk (Krasnokamensk) mine.

The EIA Report also states, that the new plant unit (LO 3) will consume approximately 20 to 40 tonnes of enriched uranium fuel per year. This equals approximately 170 to 250 tonnes of uranium concentrate (U_3O_8).

"The adverse impacts on landscape caused by uranium mining activity have been mitigated using ever more frequently the leaching of uranium directly from the ore deposit (ISL, in-situ leaching)" (FORTUM 2008, 163).

On October 16th 2007, the Ministry of Trade and Industry gave its statement on the EIA programme of the project. Within the statement, a summary of comments, opinions and statements from various ministries, experts, authorities and others is given. Among those, the Finnish Association for Nature Conservation, the Swedish environmental authority Naturvårdsverkets, the international WWF and several other organisations maintained that the impact assessment should be enhanced by considering the entire life cycle of the project, including the environmental impact of processing and transporting uranium (FORTUM 2008, Appendix 1).

6.2.2 Discussion

Environmental impact of uranium ore mining and mining waste

An important fact is, that the 170 to 250 tonnes of raw uranium needed for the production of some 20 to 40 tonnes of isotope-enriched uranium for one unit equal some 15.000 to 30.000 tonnes of ore containing 1% uranium, which first has to be mined, then transported from the mine to the enrichment plant, and last but not least deposited. As the Russian mine the ore grad is only 0.3–0.4% and will therefore produce about 3 times the waste. The transport generates large amounts of CO_2. The final disposal of uranium mining waste is probably the biggest problem of uranium mining: As discussed in (WENISCH et al. 2007), the Uranium extraction itself generates 80% of today's radioactive waste (by mass). The amount of radioactive tailings left behind in the uranium mine area is of corresponding volume. For example, the affected regions of New Mexico (USA) and Wismut (former GDR) must cope with more than 100 million tons of radioactive waste from uranium extraction on the surface (BMLFUW 2007).

The largest part of the uranium used at LO 3 nuclear power plant will be supplied by Russia's Krasnokamensk mine. Uranium mining yields huge environmental impacts. As reported in the International Herald Tribune in 2006, scientists found "dangerously high levels of radon gas emanating from the cellars of scores of houses in Oktyabrskoye, mounting levels of uranium dust and residues of heavy metals like mercury" (BELTON 2006). Oktyabrskoye was built as temporary housing for the first group of geologists who arrived at Krasnokamensk and discovered the deposits in 1964. Furthermore, the scientists stated, that the entire village was in a zone of acid pollution. "'We live in the middle of an industrial zone', said Yekaterina Zimniyeva, a former mine worker who helped build the mines in the late 1960s. 'No one should be living here.' Zimniyeva had her young grandson, who, she said, constantly suffers from chest colds. ‘Here, we eat uranium, we drink uranium, we breathe uranium,’ she said. ‘Everyone's legs here hurt terribly. People suffer heart problems, and there isn't anything to breathe’” (BELTON 2006).

During mining, uranium is removed from geological deposits that usually are geochemically stable. Residual uranium and all the separated decay products are left at the site and stored on the surface in form of dumps or as mud in simple basins. The waste products of uranium mining contain hazardous substances like thorium-230 with a half-life of 77.000 years. Thorium decays to radium and gaseous radon. "The isolation periods required for final disposal of these wastes are comparable to those of wastes from the operation of nuclear power plants. But in this case, geological storage is not taken into consideration due to the large amount of material." (BMLFUW 2007).

At the beginning, the main mining method at the Krasnokamensk mine was open pit mining, producing millions of tonnes of radioactive tailings waste. Eighty percent of the radioactivity of the original ore remains in the tailings, as well as a range of other toxic materials.

The EIA Report states, that uranium mining activity causes adverse impacts on landscape. These impacts “have been mitigated using ever more frequently the leaching of uranium directly from the ore deposit (ISL, in-situ leaching)” (FORTUM 2008, 163). It is a fact that at Krasnokamensk nowadays the extraction is carried out mainly in underground mines (FORTUM 2008, 164). On the other hand the EIA Report states, that “underground leaching is not completely without risk to the environment. When using the method, attention must be paid not to let the chemicals contaminate groundwaters” (FORTUM 2008, 163/164).

In-situ leaching (ISL)

When the uranium mining companies make use of in-situ leaching (ISL) mining technique, it means pumping acid into an aquifer in order to dissolve the uranium ore and other heavy metals and pump the solution back to the surface. The separation of the small amount of uranium is performed at the surface. "The liquid waste – which contains radioactive particles, heavy metals and acid – is simply dumped in groundwater. Inert and immobile in the ore body, the radionuclides and heavy metals are then bio available and mobile in the aquifer." (BNI 2006). Recovering of ISL mining sites is a costly operation which can be observed at the Czech uranium mine Diamir.

6.2.3 Conclusion about impacts of nuclear fuel production

Uranium mining and nuclear fuel production activities appear to have an immense impact on the land where it is mined and on people living there, independent from the mining method.

The mining regions, which are often inhabited by people, have to deal with remaining dangerous substances and health problems, left by companies which sell the uranium (and other mineral resources) to industrialized countries.

If the environmental impact of mining would be taken as seriously as the impact of cultivation of plants for energy use instead for food, both activities would have to be stopped.

Various authorities and organisations, among them the Finnish Association for Nature Conservation, the Swedish Environmental Authority Naturvårdsverkets and the WWF, maintained that the impact assessment should be enhanced by considering the entire life cycle of the project. We are in complete agreement. Furthermore, renewable sources of energy and energy conservation should be reviewed as options.

7 SPENT FUEL MANAGEMENT

7.1 Interim Storage of Spent Nuclear Fuel

7.1.1 Treatment in the EIA Report

Regarding interim storage of spent nuclear fuel, the EIA Report states, that "the spent nuclear fuel of the new power plant will be cooled initially and stored for a few years in water pools at the power plant unit. Then it will be placed in the interim storage in cooled water pools for decades – until it is disposed of at Olkiluoto, Eurajoki. The realisation of the new power plant unit requires the existing spent fuel storage facility to be expanded or building a new." (FORTUM 2008, 7).

7.1.2 Discussion

The quantities of radioactive waste in storage or disposed of at the end of 2007 are shown in the figure 1 (STUK 2007). Spent fuel from existing Loviisa NPP sum up 428 tonnes HM (heavy metal), low and intermediate level waste, excluding activated metal waste, from existing Loviisa NPP sum up 3,100 m^3 (equal to 18 TBq).

Table 13: Quantities of radioactive waste in storage or disposed of at the end of 2007 (STUK 2007).

Waste type	Facility	Quantity (Activity)
Spent fuel from NPPs	Loviisa NPP Olkiluoto NPP Total	428 tonnes HM 1,197 tonnes HM 1,625 tonnes HM
LILW from NPPs (excluding activated metal waste)	Loviisa NPP Olkiluoto NPP Total	3,100 m^3 (18 TBq) 6,100 m^3 (65 TBq) 8,600 m^3 (83 TBq)
Small user waste	Central storage	51 m^3 (23 TBq, mostly tritium)

Partially divergent to the EIA Report, the STUK report on the radioactive waste management programme in Finland from 2007 states: "As regards interim storage, extension of the storage facility for spent fuel has recently been completed the Loviisa NPP. At the Olkiluoto NPP, additional interim storage capacity is needed by 2014." (STUK 2007). It is not clear, whether the extension recently completed in Loviisa is sufficient for the realisation of the new power plant unit.

The EIA Report states, that "the fuel is usually stored in water pools but air-cooled dry storage is also possible. ... The duration of interim storage can be even 60 years if the discharge burnup is high." (FORTUM 2008, 167).

There is strong evidence, that wet intermediate storage of spent fuel is not the best solution for long-term storage, since there is evidence that fuel bundles stored in a spent fuel pool over a long period are much more difficult to handle with than fuel bundles stored in a dry storage cask. Moreover, "the potential consequences of an accident or terrorist attack on a dry cask storage facility are lower than those for a spent fuel pool: There is less fuel in a dry cask than in a spent fuel pool and therefore less radioactive material available for release." (NRC & COMMITTEE 2006).

„Radioactive material releases from a breach in a dry cask would occur through mechanical dispersion. Such releases would be relatively small. Certain types of attacks on spent fuel pools could result in a much larger dispersal of spent fuel fragments. Radioactive material releases from a spent fuel pool also could occur as the result of a zirconium cladding fire, which would produce radioactive aerosols. Such fires have the potential to release large quantities of radioactive material to the environment." (NRC & COMMITTEE 2006).

7.1.3 Conclusion about interim storage of spent nuclear fuel

For a long-term storage the pool is not an optimal technology. Critical aspects are the integrity of the fuel rods and their handling after several decades in the pool. According to the EIA Report a further extension of the interim fuel storage is envisaged in order to prepare place for the fuel from LO 3. Since it is planned to store the spent fuel in the interim storage over several decades up to 60 years, the disadvantage of the storage pool compared to a dry one should be considered. Furthermore an assessment of the risk of accidents caused by external impacts to the pool storage should be given.

7.2 FINAL DISPOSAL OF SPENT NUCLEAR FUEL

7.2.1 Treatment in the EIA Report

The final repository of spent fuel from TVO's and Fortum's NPPs is intended to be located 400–500 m underground in the bedrock at Olkiluoto. An EIA concerning this project was completed in 1999. After a positive decision in principle Posiva Oy, a company own by the operators and responsible for the spent fuel management, started preparation of an underground research facility called ONKALO at Olkiluoto. The objective of this project is to obtain detailed information concerning the bedrock for the purpose of designing the disposal facility and assessing its safety. The spent fuel will be packed into airtight metal canisters before being transferred into the repository. Posiva intends to supply the application for a construction licence for the Spent fuel repository by the end of 2012. The disposal of spent fuel into the repository is scheduled to start in 2020.

The EIA Report states that according to the assessment of the impacts of the final disposal of spent nuclear fuel performed in 1997–1999 the environmental impacts are small: “Impacts on nature and use of natural resources are small and limited to the immediate vicinity of the power plant. Except for any psychosocial effects, the project has no significance for people’s health. The project has positive socioeconomic impacts.” (FORTUM 2008, 169).

The environmental impacts of the extended repository will be considered in the EIA carried out by Posiva in 2008 (FORTUM 2008, 169).

“In the final disposal conditions, the copper canister withstands corrosion for a very long time. The clay surrounding the canister protects it from any small movements of the earth’s crust, and prevents very efficiently the flowing of water in the final disposal rooms.” (FORTUM 2008, 168).

“When the copper canister finally loses its integrity the radioactive materials released from the fuel migrate poorly through the clay buffer. When they permeate the clay buffer, they remain in the minerals of the bedrock and are diluted with the groundwater in the bedrock.” (FORTUM 2008, 168).

The EIA Report states that the safety of the final disposal of spent fuel is based on the technical and natural multiple barriers. Furthermore, the long-term safety of the final repository is assessed by making a very conservative safety analysis and justification. “According to the results, it can be assumed that the adding of the fuel from the new power plant units does not change the situation noticeably, although the quantity of fuel to be finally disposed of ... would increase significantly.” (FORTUM 2008, 170).

7.2.2 Discussion

First of all, copper will withstand corrosion for a longer time span only if the conditions in the final repository are almost totally void of oxygen. Research results indicate that hundreds of meters down in the bedrock of Olkiluoto, nowadays the groundwater is virtually void of oxygen and flows very slowly. Moreover, the clay surrounding the canister prevents the flowing of water in the final disposal rooms. Hence the corroding effect on the canisters and the spent nuclear fuel, and the transport of radioactive substances away from the bedrock to environment and living nature, are believed to be insignificant.

There are strong indications, that this cannot be ensured. An international exercise resulted in the following dose impact of deep spent-fuel disposal in a granite environment after more than 10.000 years: "The impact is zero in the first 10.000 years following sealing of the facility. Then highly mobile iodine-129 is the first isotope to reach the outlet and contributes most to the dose. After several hundreds of thousands of years, the heavy atoms (226Ra, 230Th) from decay chains 4N[9] (232Th chain), 4N+1 (241Am and 237Np chain), 4N+2 (238U chain) and 4N+3 (235U chain) take over the running." (BONIN 2002).

[9] “N” means the number of nucleons. This system is used to identify the different radioactive decay chains.

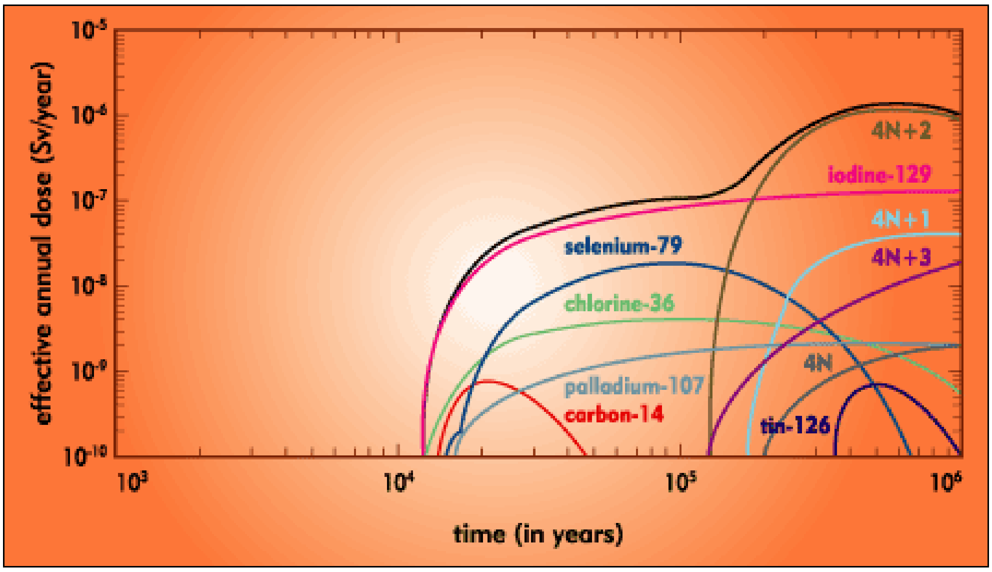

Figure 2: The dose impact of deep spent-fuel disposal in a granite environment (BONIN 2002).

Since final disposal of spent nuclear fuel is a matter of time-span of several hundreds of thousands of years, even a very slow water flow towards the canisters means also a (very slow) flow away from it. Even this slow water stream is able to carry radioactive substances away from the bedrock surrounding potential damaged canisters – in case of any unforeseen circumstances such as stronger corrosion of canisters due to water break-in and coincident changes in oxygen content of the groundwater – and could result in a strong impact to environment and living nature.

At present Finland investigates the bedrock at Olkiluoto in order to locate the repository there. This is assumed to be ready for disposal in 2020. There is no discussion in the EIA Report of alternatives if the bedrock reveals to be not adequate for long-term safety. The safety analysis for deep geological repositories contains large uncertainties, in particular regarding the long-term function of technical and geological barriers.

7.2.3 Conclusion about final disposal of spent nuclear fuel

The EIA Report states, that the long-term safety of the final repository has been assessed by making a very conservative safety analysis and justification. According to the state of science and technology final waste disposal for spent fuel can be realized in deep geological formations. Great advances have been made concerning the accomplishments of safety analysis. The deep geological repository of high level waste can be rated as the most safe way to treat radioactive waste. Anyway there are large uncertainties in the results of the safety analysis. The assessment of the influence of ice-ages with fault movements, land uplift, earthquakes and the creation of new weakness zones also contributes to these uncertainties.

A very long storage period is required. In particular, the long term capability of the technical and geological barriers cannot be guaranteed for such long periods.

There is no discussion in the EIA Report of alternatives if the bedrock reveals to be not adequate for long-term safety.

8 ALTERNATIVES AND ZERO OPTION

8.1 Treatment in the EIA Report

Alternatives are discussed in chapter 15 of the EIA Report. Regarding the NPP boiling or pressurized reactors are mentioned with a bandwidth of 1,000 MW to 1,800 MWe. For cooling water four options are presented: local intake and discharge, local intake and remote discharge, remote intake and local discharge and remote intake and remote discharge. The discharge water will increase the sea temperature in all options, also the nutrient load will increase in the vicinity of the cooling water discharge location (FORTUM 2008, 177).

The planned NPP will have an estimated total efficiency of 35–40% (FORTUM 2008, 41). This efficiency could be improved up to 60% by also producing heat. If the plant can produce 4,600 MW thermal power, the electric capacity would decrease from 1,800 MW to 1,560 MW, and 1,200 MW would be available for district heating. Fortum states that “in practice there are consumers for the 1,200 MW of district heat only in the metropolitan areas.” (FORTUM 2008, 41).

The discussion of environmental impacts of the zero option compares nuclear energy to the production of the same amount of electricity by other projects, without discussing location or production technology of these projects (FORTUM 2008, 138). Several fuels (coal, water, wood, natural gas, peat, oil, waste, wind, solar energy and others like fusion power or wave energy) are discussed regarding their contribution to Finland´s electricity needs for today and their general potential for the future. Also import of electricity is mentioned. Fortum states that “[i]f the project will not be implemented, the amount of electricity corresponding to the production of the new power plant unit will be probably produced by a combination of fossil fuels, peat and renewable energy sources.” (FORTUM 2008, 140).

In comparison to the other fuels in discussion Fortum argues that “[n]uclear power production will not cause carbon-dioxide, sulphur dioxide, nitrogen oxide or particle emissions.” (FORTUM 2008, 140).

If LO 3 will not be implemented, “Fortum wants to keep the area in Hästholmen for later additional building of nuclear power” (FORTUM 2008, 137).

If the NPP will not be constructed, positive employment effects will not be realised. This will be the most significant impact of non-implementation of the project (FORTUM 2008, 177).

The costs of the project and its alternatives are not described in the EIA Report.

8.2 Discussion

Fortum´s statement that all alternatives for cooling water will lead to an increase of the thermal load is an important environmental impact. The City of Loviisa adds in its Statement on the EIA Assessment Program that accelerated growth of aquatic plants has already been observed (MTI 2007, 203). It seems that more effort should be put in mitigation of the effects of the thermal load of the cooling water.

No alternative options for electricity production or options for investment in energy efficiency are discussed in the EIA Report.

In the statement on the EIA Assessment Program several Finnish environmental organisations stated that energy-saving options should be reviewed, the need for the project should be justified more sufficiently, and equal weight should be given to the alternative options for satisfying the Finnish energy need. Also foreign environmental authorities (Sweden, Estonia) argue that alternative energy production should be included in the assessment (MTI 2007, 207f.).

Regarding these criticisms the MTI declared in its Statement of the Contact Authority that "the organisation responsible for the project is a company that sells electricity. Therefore, it cannot access any significant means of energy conservation or operational efficiency and has limited opportunities to influence the electricity use of its customers." But the Government itself "will include information on energy conservation and efficiency. However, this perspective will cover the Finnish energy supply as a whole and thus could not be applied to the issue of replacing the power plant under review." (MTI 2007, 210).

During the consultation of Austria and Finland in May 2008 a presentation of STUK addresses the Finnish energy policy (PÖLLÄNEN 2008): The new Finnish Climate and Energy Strategy is not ready by now, it will be dealt with in Parliament in autumn, 2008. The decision-on-principle on new nuclear power plants probably depends on the decision on the energy strategy.

To use a nuclear power plant for production of combined heat and electricity is questionable because NPPs are usually located in remote areas because of their risk potential. Therefore it is questionable if there are enough consumers for the produced heat. From the point of efficiency and sustainability gas-powered combined plants located near the consumers would be a better solution, especially if not only natural gas but also bio-gas could be used.

Regarding the statement of Fortum that nuclear energy will not cause emissions of CO_2, it has to be stated that nuclear electricity production is not CO_2 free if the whole uranium fuel cycle is taken into consideration (WENISCH et al. 2007). Using current uranium ore grades (~ 2% concentration) results in 33 g of CO_2 equivalent per kWh of nuclear electricity in Germany. In France, it is only 8 g, while it is higher in Russia and in the USA, 65 g and 62 g respectively. Fortum uses only uranium from Russia for the Loviisa plant and creates therefore CO_2 emissions of 65 g/kWh.

One reason for this is the quality of uranium ore: the lower the grade, the more CO_2. A substantial increase of nuclear electricity generation would require the exploitation also of lower grade uranium ores and thus would increase the CO_2-emissions up to 120 g, which is more than other energy technologies: natural gas co-generation 50–140 g, wind power 24 g, hydropower 40 g, energy conservation 5 g CO_2 eq/kWh_{el} (FRITSCHE 2007). With 65 g CO_2 per kWh is the uranium from Russia all but not CO_2 free and not better than a modern gas-fired plant.

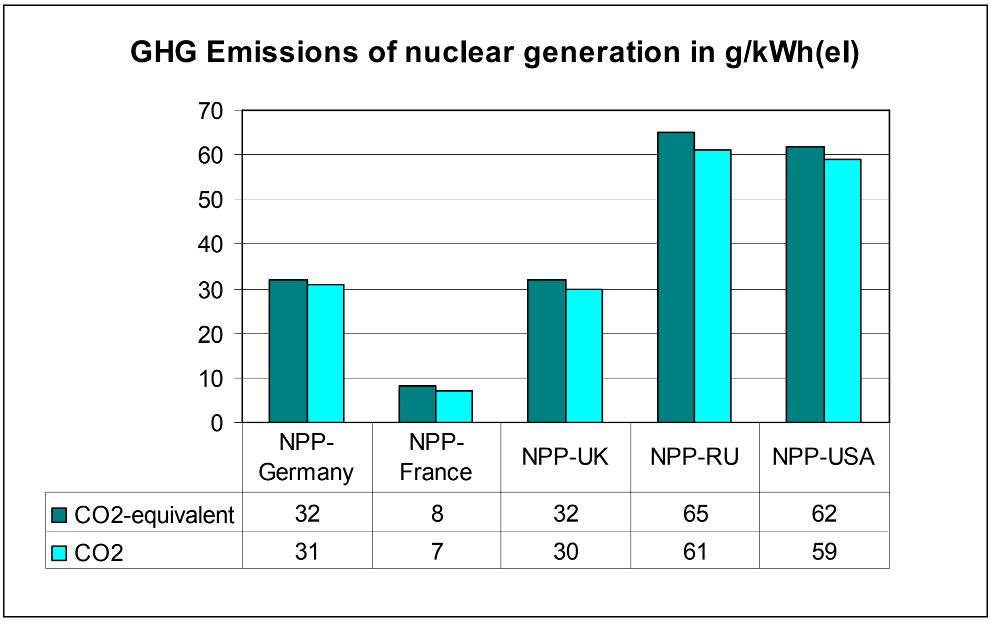

Figure 4: Greenhouse gas Emission of Nuclear Generation (FRITSCHE 2007).

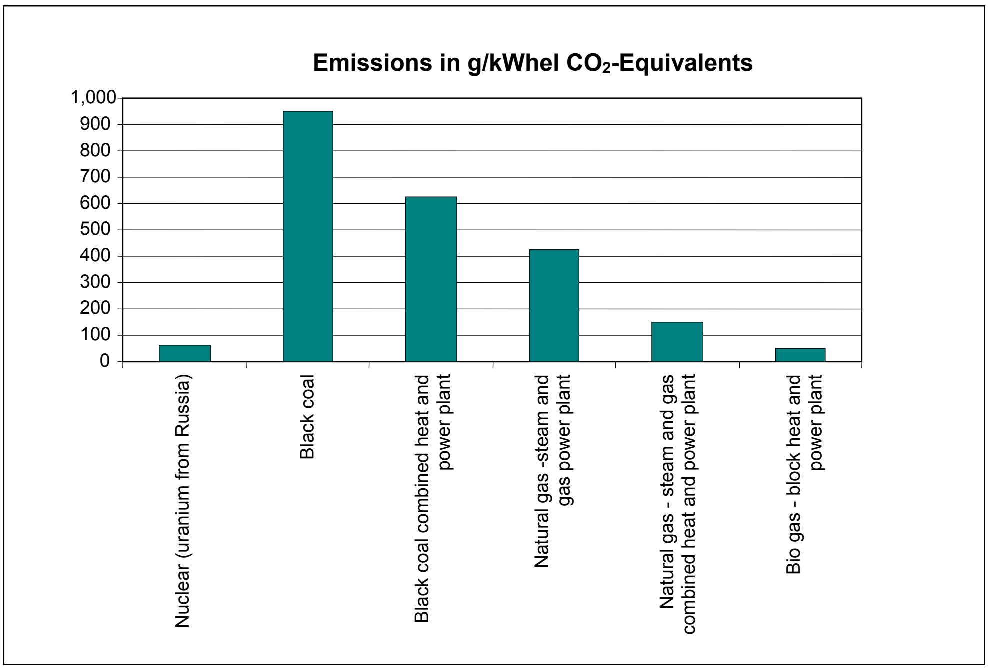

Figure 3: CO_2 Equivalents of different energy sources (FRITSCHE 2007).

Also the costs of the project and its alternatives are missing. For a realistic cost assessment it would be necessary to include the whole costs of the fuel chain.

8.3 Conclusion

In the EIA Report the discussion of alternative energy solutions is very narrow. It is difficult to assess if a new nuclear power plant is really needed in Finland if data about energy need and future trends are not included in the EIA Report.

To use a nuclear power plant for production of combined heat and electricity is questionable because NPPs are usually located in remote areas because of their risk potential. Therefore it is questionable if there are enough consumers for the produced heat. From the point of efficiency and sustainability gas-powered combined plants located near the consumers would be a better solution, especially if not only natural gas but also bio-gas could be used.

If the total nuclear chain and the management of all types of radioactive waste are considered, nuclear energy cannot be regarded as CO_2 free and environmentally sound. In particular uranium mining and fuel fabrication cause a significant impact on the environment (see also chapter 6 of this statement). A complete comparison of the real costs and risks of nuclear energy production with those of renewable energy production alternatives proves that there is no advantage of nuclear electricity towards renewable energy (BMLFUW 2007). With 65 g/kWh electricity from nuclear power production is all but CO_2 free.

Nuclear energy also proves to be a comparatively costly measure to reduce CO_2 emissions. Energy efficiency measures, renewable energies and alternative solutions in the wider sense replace 2.5 to 10 times as much CO_2 per unit investment (BMLFUW 2007).

9 QUESTIONS

Procedure

1. Why is the information about the reactor types in discussion not made public by the applicant, especially as in a similar UK procedure the availability of comprehensive information is obviously possible?
2. Why did Fortum neglect the requirements of the MTI's statement of the scoping stage of the EIA to provide a review of current power plants on the market which are suitable for the project under review?
3. Since - beside limited emission targets - no detailed safety requirements seem to be published, could STUK provide a comprehensive list of the specific safety requirements for Generation III reactors in Finland for orientation?
4. Will the Ministry demand the fulfillment of issues required in the MTI's statement of the scoping stage of the EIA?
5. Is there a timetable for presenting missing information to the public and to the ESPOO partners?
6. Is the understanding correct, that the decision-in-principle will be made in spring 2009, after the new Finnish Climate and Energy Strategy will have been dealt with in Parliament in autumn 2008?

Reactor types

1. The reactor types listed in the EIA Report as being in non-binding consideration for LO 3 are in different stages of development; most of them are still at the design stage, only one of them (ABWR) has operational experience. Especially the accuracy and reliability with which the hazards can be assessed will also vary considerably and should therefore be described in detail. How will the Finnish authorities address this circumstance during the following decision process?
2. Which criteria has Fortum defined for the selection of the reactor? Can they be described and reported to the public before a governmental decision is taken?

Questions concerning the EPR

1. Does the Finnish side agree to the assessment that the main advantage of the measures for core melt mitigation implemented at the EPR appears to be that filtered venting can be avoided; and that the core catcher yields no significant improvement regarding the probability of large releases, compared to modern Generation II PWRs?
2. For Olkiluoto 3, CDF from internal events is about 1.44E-6/yr, more than twice as high as the results for the UK-EPR. The CDF for internal events does not depend significantly on the site selected; how can this discrepancy be explained – different PSA methodology, additional safety measures to be implemented at the UK-EPR?
3. Does the Finnish probabilistic target of 5E-7/yr for large releases cover all accident initiators – internal events, internal hazards and external events?

4. Are there more detailed PSA results for Olkiluoto 3 available today, compared to the state of 2005, particularly concerning containment behavior, including internal and external hazards?
5. What is the status of the analyses performed for Olkiluoto 3 regarding the crash of a commercial airliner, including analyses of vibrations? Are there more results available today, compared to the state of 2005?
6. 2A-LOCA is a design basis accident for the Olkiluoto 3 EPR, as opposed to Flamanville and the UK-EPR, where it is dealt with in the context of risk reduction. Regarding the analyses supporting this statement – were all parameters and assumptions entering them selected in a conservative manner (regarding output of safety systems, thermo-hydraulic assumptions, methodology used for thermo-hydraulic safety analysis, operational state of the reactor etc.)?

Questions concerning the ESBWR:

1. Does the Finnish side agree that passive safety can have drawbacks regarding flexibility of reaction in critical situations – particularly in case of high power density?
2. Does the Finnish side agree that the core catcher concept of the ESBWR, without prior melt retention, might lead to specific problems (in addition to the general uncertainties involved in core catcher functioning)?
3. The CDF values published for the ESBWR are low, particularly when compared, e.g., to those for the EPR. How meaningful are the ESBWR CDF values in the view of the Finnish side, taking into account the relatively early stage of development of the ESBWR and the fact that PSA results in the order of 1E-08 or 1E-09/yr are beset with particularly high uncertainties?
4. To the knowledge of the authors of this statement, no results of level 2 PSAs, particularly no values for large release frequencies, have been published for the ESBWR. Does the Finnish side have such results?
5. What is the assessment of the Finnish side of the general vulnerability of the ESBWR to external hazards, particularly to attacks?

Safety and accidents

1. Why has Fortum ignored the requirement of the MTI in its statement on the scoping procedure of the EIA to present the safety planning criteria of the prospective plant with respect to the limitation of radioactive emissions as well as an assessment of how the safety requirements in force will be met?
2. Concerning the source term, what is the relation between EUR Criteria and the Finnish Regulation for Limited Impact?
3. Method and input data for the dose assessment are not explained in the EIA Report. For the dose assessment three computer models are used; please, provide a description of the dispersion models and the weather data used for the assessment. Is one of these programs appropriate for modeling long-range transport and dispersion of radionuclides?

4. Concerning the large bandwidth of 2 orders of magnitude of CDF and LRF assessments for the reactors under review, will the release frequency and the related emissions (source term) be criteria for the selection of the reactor?

Spent fuel management

1. The interim storage of spent fuel in a pool over long periods seems not to be an optimal solution. An enlargement of the interim storage is envisaged. Considering the disadvantages of wet storage, would dry storage not be a safer option?

Alternatives and Zero Option

1. Are there potential heat consumers near the NPP? Wouldn´t it be a better solution to use i.e. gas-powered combined plants that can be located near the consumers?

10 REFERENCES

AREVA (2005): AREVA – Framatome ANP: EPR; Brochure, March 2005.

AUA – Australian Uranium Association(2008): Advanced Nuclear Power Reactors; Nuclear Issues Briefing Paper 16, March 2008.

BÄVERSTAM, U. (1996): Lena 95. A program for real time decision support. User's manual. Swedish Radiation Protection Institute. 1996.

BEARD, J.A. (2007): ABWR Safety – PRA, Containment Response & Severe Accidents; GE Energy/Nuclear, Presentation, April 2007.

BELTON, C. (2006): For Russia, dependence on 'a man-made disaster'. International Herald Tribune. January 12, 2006. Article on www.Iht.com: http://www.iht.com/articles/2006/01/12/news/uranium.php?page=2.

BITTERMANN, D. et al. (2003): Aktuelle Neu-Entwicklungen (Gen III) mit Schwerpunkt EPR und SWR 1000; Fachtagung Fortschritte bei der Beherrschung und Begrenzung der Folgen auslegungsüberschreitender Ereignisse, Forschungszentrum Karlsruhe, Dezember 2003.

BMLFUW – Federal Ministry of Agriculture and Forestry, Environment and Water Management Nuclear Power (2007): Climate Policy and Sustainability. An Assessment by the Austrian Nuclear Advisory Board. Vienna.

BMU – Bundesministerium für Umwelt, Naturschutz und Reaktorsicherheit (2007): Übereinkommen über nukleare Sicherheit – Bericht der Regierung der Bundesrepublik Deutschland für die Vierte Überprüfungstagung im April 2008; Berlin, Oktober 2007.

BNI – Beyond Nuclear Initiative (2006): Yellowcake Country? Australia's uranium industry. Australia.

BONIN, B. (2002): Radioactive waste and its impact on the environment. Storage and/or disposal. Clefs CEA, Vol. 46.

BOSSEL, U. (2007): Das Märchen vom CO_2-freien Atomstrom. Solarzeitalter Vol. 1/2007.

BRETTSCHUH, W. & MESETH, J. (2000): SWR 1000 vor Angebotsreife; atw 45/6, Juni 2000, p. 369–373.

BRETTSCHUH, W. & SCHNEIDER, D. (2001): Moderne Leichtwasserreaktoren – EPR und SWR 1000; atw 46/8-9, August/September 2001, p. 536–541.

BRUSCHI, H.J. (2004): The Westinghouse AP1000 – Final design approval; Nuclear News, November 2004, p. 30–35.

CNS – Convention on Nuclear Safety (2002): Second Review Process; Responses to the Questions on the National Report of Germany, p. 48. Bei der Auflistung der Häufigkeiten von Gefährdungszuständen fehlen bei Unterweser als einzigem laufendem AKW die Angaben.

CSNI – OECD Nuclear Energy Agency, Committee on the Safety of Nuclear Installations(2002): Implementation of Severe Accident Management Measures; NEA/CSNI/R (2002) 12, July 2002.

DIEHL, P. (2006): Reichweite der Uranvorräte der Welt. Greenpeace Germany, Berlin.

DNE – University of Berkeley, Department of Nuclear Engineering (2008): http://www.nuc.berkeley.edu/designs/abwr/abwr.html (seen April 18, 2008).

EC (1367/2006): Regulation (EC) 1367/2006 of the European Parliament and of the Council on the application of the provisions of the Aarhus Convention on Access to Information, Public Participation in Decision-making and Access to Justice in Environmental Matters to Community institutions and bodies ("Aarhus Regulation").

EC (2003/35): Directive 2003/35/EC EC of the European Parliament and of the Council of 26 May 2003 providing for public participation in respect of the drawing up of certain plans and programmes relating to the environment and amending with regard to public participation and access to justice Council Directives 85/337/EEC and 96/61/EC.

EC (2003/4): Directive 2003/4/EC of the European Parliament and of the Council of 28 January 2003 on public access to environmental information and repealing Council Directive 90/313/EEC.

EC (97/11): Council Directive 97/11/EC of 3 March 1997 amending Directive 85/337/EEC of 27 June 1985 on the assessment of the effects of certain public and private projects on the environment.

EDF – Electricité de France (2006): Rapport Préliminaire de Sûreté de Flamanville 3, version publique; 2006.

EIA Act (2006): Ministry of the Environment, Finland: Act on Environmental Impact Assessment Procedure (468/1994, amendments up to 458/2006 included), unofficial translation, http://www.ymparisto.fi/download.asp?contentid=84193&lan=en.

ESPOO Convention (1997): Convention on Environmental Impact Assessment in a Transboundary Context. 201.Übereinkommen über die Umweltverträglichkeitsprüfung im grenzüberschreitenden Rahmen samt Anhängen und Erklärungen, ausgegeben am 28. November 1997, Teil III, BGBl für die Republik Österreich.

EUR – European Utility Requirements for LWR Nuclear Power Plants (2001): Volume 2, Generic Nuclear Island Requirements, Chapter 1, Safety Requirements; Revision C, April 2001

EWG – Energy Watch Group (2006): Uranium Resources and Nuclear Energy. Background paper prepared by the Energy Watch Group. EWG-Series No 1/2006. Achen.

FORTUM (2008): Fortum Power and Heat Oy: Environmental Impact Assessment Report. Supplementing the Loviisa Nuclear Power Plant with a Third Plant Unit.

FRITSCHE, U. R. (2007): Treibhausgasemissionen und Vermeidungskosten der nuklearen, fossilen und erneuerbaren Strombereitstellung. Arbeitspapier. Öko-Institut e.V. – Institute for Applied Ecology, Darmstadt.

GENERALOV, V. (2007): More Russian Evolution; Nuclear Engineering International, October 2007, p. 20–21.

HINDS, D. & MASLAK, CH. (2006): Next-generation nuclear energy: The ESBWR. nuclear news, January 2006, p. 35–40.

HIRSCH, H. (2006): Hazards of New Reactor Types; presentation at Seminar “Kernenergie 'Ja, mits' of 'Nee, tenzij'”, wise Amsterdam, November 9, 2006.

HÖRMAYER, N.; MRAZ, G. & WENISCH, A. (2007): Participation in EU policy – A guide for NGOs and citizens. Legal and political options to participate in antinuclear energy policy. Set 04.

HUMBER, Y. (2008): Uranium Holding May Spend $8.6 Billion in Russia. Article on www.bloomberg.com. http://www.bloomberg.com/apps/news?pid=20670001&refer=home&sid=aj0iS8fi8m8k.

IAEA – International Atomic Energy Agency (1992): Pilot studies on management of ageing of nuclear power plants, Results of Phase I; IAEA-TECDOC-670, Vienna, October 1992.

IAEA – International Atomic Energy Agency (2004): Status of advanced light water reactor designs 2004; IAEA-TECDOC-1391, May 2004.

MTI (2005). Outline of the Energy and Climate Policy for the Near Future – National Strategy to Implement the Kyoto Protocol. Government Report to Parliament 24 Nov 2005.

MTI (2007): Environmental Impact Assessment Programme for the Loviisa-3 Nuclear Power Plant Unit: Statement by the Contact Authority. Appendix in FORTUM (2008, 196–215).

NEA (2002): OECD Nuclear Energy Agency: Advanced Nuclear Reactor Safety Issues and Research Needs; Workshop, Paris, February 18–20 2002.

NEI (2007): Nuclear Energy Institute (NEI) Nuclear Notes, March 14, 2007. http://neinuclearnotes.blogspot.com/2007/03/txu-to-order-mitsubishis-apwr.html (seen April 18, 2008).

NEIMAG (2007): Nuclear Engineering International (Magazine), September 2007, p. 5.

NORDLUND, G.; ROSSI, J.; SAVOLAINEN, I. & VALKAMA, I. (1985): (1985): Ilmaan joutuvien radioaktiivisten päästöjen vaikutus väestön säteilyannoksiin laajoilla alueilla. VTT. Tiedotteita 525. Espoo 1985.

NRC – National Research Council & Committee on the Safety and Security of Commercial Spent Nuclear Fuel Storage (2006): Safety and security of commercial spent nuclear fuel storage: public report. http://www.nap.edu/catalog.php?record_id=11263.

OLSON (1997): Olson, S. & Haukeland, S.: Light Water Reactors for the Next Century; 21st Annual International Symposium of the Uranium Institute, 1997. http://www.world-nuclear.org/sym/1997/olsson.htm (seen April 16, 2008).

PÖLLÄNEN, L. (2008): Nuclear power plant project – Regulatory aspects and safety criteria. Consultations between Finland and Austria in accordance with article 5 concerning the extension of the Olkiluoto nuclear power plant by a fourth unit. 26 May 1008.

SAALEM, A. (2000): WMC: Olympic Dam Mine. The 2000 Aboriginal Management Case Writing Competition. Australia.

SAIKKONEN, T. (1992): Radioaktiivisesta päästöstä ihmiselle aiheutuvan säteilyannoksen arviointi. Master's Thesis HTTK. 1992.

SEHGAL, B.R. & HYUN SUN PARK (2004): Pre-Project on Development and Validation of Melt Behavior in Severe Accidents; Nordic nuclear safety research, NKS-99, June 2004.

SEVON, T. (2005): Molten Core-Concrete Interactions in Nuclear Accidents: Theory and Design of an Experimental Facility; Master's Thesis, Helsinki University of Technology, Department of Engineering, Physics and Mathematics, October 2005.

SSK – Strahlenschutzkommission des Bundesministeriums für Umwelt, Naturschutz und Reaktorsicherheit (2004): Leitfaden für den Fachberater Katastrophenschutz der Katastrophenschutzleitung bei kerntechnischen Notfällen; Berichte der Strahlenschutzkommission (SSK), Heft 37, 2004.

STATE 395/91: Decision of the Council of the State on the general regulations for the safety of nuclear power plants.

STOHL, A.; HITTENBERGER, M. & G. WOTAWA (1998): Validation of the Lagrangian particle dispersion model FLEXPART against large scale tracer experiments. Atmos. Environ. 32, 4245-4264. http://zardoz.nilu.no/~andreas/flextra+flexpart.html.

STUK – Finnish Radiation and Nuclear Safety Authority (2005): Safety Assessment of the Olkiluoto 3 Nuclear Power Plant Unit for the Issuance of Construction License. 2005.

STUK – Finnish Radiation and Nuclear Safety Authority (2007): Radioactive waste management programmes in OECD/NEA member countries. Finland. National nuclear energy context. http://www.stuk.fi/ydinturvallisuus/ydinjatteet/ydinjate/en_GB/ydinjate/_files/73810475841424335/default/fin_country_profile.pdf

TSUJI (1998): A. Tsuji et al.: Completion of ABWR Plant – Kashiwazaki-Kariwa Nuclear Power Station Unit Nos. 6 and 7; Hitachi Review 47/5, 1998, p. 157–163.

TVO (2008): Environmental Impact Assessment Report. Finland.

UK-EPR (2008): AREVA & EDF: EPR Safety, Security and Environmental Report for the Fundamental Safety Review in the UK. http://www.epr-reactor.co.uk/scripts/ssmod/publigen/content/templates/show.asp?P=139&L=EN, seen May, 2008.

UK-ESBWR (2008): GE–Hitachi: Safety, Security and Environmental Report – Preliminary Safety Report; Rev. 3, ESBWR Design Control Document, Tier 2. www.gehgenericdesignassessment.co.uk/document_download.jsp, seen May, 2008.

WEC – Westinghouse Electric Company (2007): AP1000–Simple, Safe, Innovative; 2007

WENISCH, A.; KROMP, R. & REINBERGER, D. (2007): Science or Fiction. Is there a Future for Nuclear? Austrian Institute of Ecology, Vienna.

WENISCH, A.; HIRSCH, H.; KROMP, R. & MRATZ, G. (2008b): NPP Olkiluoto-4. Expert Statement to the EIA Report. Ordered by the Federal Ministry for Agriculture, Forestry, Environment and Water Management, Dept V/6. Report, REP-0162. Umweltbundesamt, Wien.

WENISCH, A. & KROMP, R. (2008): Scoping Phase of the EIA Program for NPP Olkiluoto-4, Expert Statement.

WENISCH, A. & KROMP, R. (2008b): Fennovoima Oy Scoping Phase of the EIA Program for a NPP. Expert Statement. Vienna.

WISE (2001): Coeytaux, X.; Faïd, Y.; Marignac, Y. & Schneider, M. (2001): La Hague Particularly Exposed to Plane Crash Risk; WISE-Paris Briefing NRA-v4. http://www.wise-paris.org/english/ourbriefings_pdf/010926BriefNRA1v4.pdf.

WNA – World Nuclear Association(2008): Nuclear Power in the USA. Information Papers, April 2008.

WNIH – Nuclear Engineering International (2007, 2008): World Nuclear Industry Handbook, 2007 and 2008 editions.

11 GLOSSARY

ABWR .... Advanced Boiling Water Reactor

Al........................ Aluminium

Am...................... Americium

AP .... Advanced Passive

APR.................... Advanced Power Reactor

APWR .... Advanced Pressurized Water Reactor

ARMZ................. Russian state-owned mining company

ATWS................. anticipated transients without scram

BDBA .... Beyond Design Basis Accident

BiMAC................ Basemat-internal Melt Arrest and Coolability

BNFl .... British Nuclear Fuels plc

BWR................... Boiling Water Reactor

CCFP .... Conditional Containment Failure Probability

CD .... Core Damage

CDF.................... Core Damage Frequency

CFS.................... Cavity Flooding System

CHRS................. Containment Heat Removal System

Cs....................... Caesium

DBA.................... Design Basis Accident

EC .... European Commission

ECCS .... Emergency Core Cooling System

EIA .... Environmental Impact Assessment

EPR.................... European Power Reactor

ERF .... Early Release Frequency

ERVCS............... External Reactor Vessel Cooling System

ESBWR.............. Economic Simplified Boiling Water Reactor

EU .... European Union

EUR .... European Utilities Requirements

FKE .... accident release category, where 6% of the core melts

Fortum................ Fortum Heat and Power Oy

GDCS................. Gravity Driven Cooling System

GE...................... General Electric

GHG................... Greenhouse Gases

GKN-2 .... Neckarwestheim 2 nuclear reactor unit

HM .... Heavy Metal

I .... Iodine

IAEA................... International Atomic Energy Agency

ICRP .... International Commission on Radiation Protection

IEA .... International Energy Agency

IRWST................ in-containment refuelling water storage tank

ISL...................... In-situ leaching

LILW................... Low and Intermediate Level Waste

LOCA .... Loss of Coolant Accident

LO 3 .... Loviisa Unit 3

LOOP .... Loss of Offsite Power

LRF .... Large Release Frequency

LWR .... Light Water Reactor

MCR................... Main Control Room, Master Control Room

MEE .... Ministry of Employment and the Economy (former MTI)

Mg .... Magnesium

MIPS .... Material Input Per Service Unit

MSLB .... Main Steam Line Brake

mSv.................... Milli Sievert

MTI..................... Ministry of Trade and Industry, since Dec 2007: Ministry of Employment and the Economy (MEE)

MW..................... Megawatt

MWe................... Megawatt electric

NGO................... Non Governmental Organisation

Np .... Neptunium

NPP.................... Nuclear Power Plant

NRC .... Nuclear Regulatory Commission (USA)

OL-4 .... Olkiluoto Unit 4

OPP .... Overpressure Protection

PCC .... Plant Condition Categories

PCCS .... Passive Containment Cooling System

PSA.................... Probabilistic Safety Assessment

PWR................... Pressurized Water Reactor

r.yr...................... Reactor year

Ra .... Radium

Rn .... Radon

RPV.................... Reactor Pressure Vessel

RRC .... Risk Reduction Categories

SG...................... Steam Generation

SNF.................... Spent Nuclear Fuel

Sr .... Strontium

SWR................... Siedewasserreaktor, Boiling Water Reactor

Sv....................... Sievert

TBq .... Tera Becquerel

Th....................... Thorium

TVEL .... Russian nuclear fuel extraction and processing company

TVO.................... Teollisuuden Voima Oy

TWh .... Tera-Watthours, 10^{12} Wh

t/yr .... Tonnes per year

U .... Uran

VVER .... Vodo-Vodyanoy Energeticheskiy Reactor

WNA................... World Nuclear Association

yr .... Year

YVL .... Regulatory Guides on Nuclear Safety

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